BURNUP AND FEASIBILITY STUDY OF
LOW POWER DENSITY PWR'S

BY

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To Him who cares for all

and who gave us this world,

as an insignificant sample of my deep appreciation

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Abstract of Dissertation Presented to the Graduate Council
of the University of Florida in Partial Fulfillment of the
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Operational and safety problems of current Pressurized Water
Reactors are often associated with the high power density level of
the cores. An alternate use of current-design cores is proposed by
reducing the power density. The effects should be improved safety,
improved ore utilization, and improved operational characteristics.

A scoping study is performed in order to define core
parameters suitable for optimization under the low power density
characteristics, while minimizing redesign requirements.

A neutronic optimization study of the reactor cores is
performed by systematic changes in the fuel lattice pitch. A new
core burnup computational model (CRIBUR) is developed, which allows
calculation of the burnup and isotopic analysis of a multi-batch
core in its equilibrium cycle with a moderate computational and
human effort. CRIBUR provides better accuracy and sensitivity than
other known existing models of comparable scope, with a moderate

computational effort. The code is benchmarked against actual core data and against multi-dimensional diffusion theory core calculations, and its sensitivity to several of the calculational parameters is also tested.

The thermal-hydraulic behavior of the low-power cores is compared to that of the standard reactors, and their enhanced safety margins are clearly demonstrated.

The low-power cores yield higher burnup levels than the standard reactors. Ore utilization is also improved in a once-through fuel management policy. Isotopic comparisons are presented. Core cycles are drastically increased. Plant availability and capacity factors are also increased as a result of both the reduced impact of refueling downtimes and the reduced forced outage time resulting from the improvement of operational characteristics.

An economic comparison of the low-power cores is presented as a function of the core power level and the level of inflation. Low-power cores are at an economic disadvantage when compared to the standard reactor primarily because of the impact of the increased capital cost. The fuel cycle cost is also higher because of the long fuel core residence time. This economic disadvantage needs to be weighed against the improved safety and operational reliability to determine the commercial feasibility of the concept.

CHAPTER I
INTRODUCTION

1.1. Background

The most widely used reactor system for present and near future commercial production of nuclear energy is the Light Water Reactor (LWR). The LWR's were developed originally as compact, light-weight, high-power units suitable for ship propulsion and power, and they indeed performed as intended. The large research and development efforts of the U.S. military establishment for the LWR systems made them attractive for commercial power generation, and thus, they were deployed in land-based electric power plants.

LWR's have been generating commercial electricity for a considerable amount of time, totaling over 500 reactor-years of operation in the United States alone. However, the safety systems and engineering safeguards intended to prevent or to contain and reduce the impact of accidents have become extremely sophisticated and costly. This is due to several cumulative reasons: 1) the fact that LWR's work under conditions that are close to technological limits, 2) the high power density typical of these compact reactors, 3) the fuel and coolant conditions, and 4) the inherent "core-cooling" problems in case of accidents or severe transients.

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The many operational transients, safety-related incidents and the rare but significant accidents resulting in extended reactor shutdown, e.g. Three Mile Island and others, clearly substantiate this assertion.

A key dilemma appears immediately: a very high power density reactor has intrinsic serious safety problems; however, economic considerations have resulted in larger, high power density reactors working closer to their technological limits and requiring improved engineering safeguards. The industry has naturally worked towards extracting the maximum power from a given core, and also towards extending the reactor fuel cycle time in an attempt to reduce the economic impact of refueling outages and fuel carrying charges. Both these goals tend to demand performance from the reactors in a manner that is not always compatible with strict safety considerations.

Several questions need to be asked at this point. First: Is there a way of relieving the LWR's cores from these limiting situations without incurring an unacceptable economic penalty? Second: Will the new concept or approach need extensive (and thus costly and long) research and development and therefore have no impact in the short term? Third: Will the new concept be easily licensable, or will it require completely new licensing regulations and processes that would take many years to develop and create significant uncertainties for its deployment?

The first part of the first question has a relatively obvious answer from the technical point of view: reducing the power density

of the cores and making other technological fixes while maintaining the basic reactor design and technology will increase safety margins in the critical operational variables. However, the plant economics could suffer a significant penalty since a reduced power density requires a higher capital investment per unit of installed power, and this is the major part of the cost of nuclear-produced energy. On the other hand, the reduction of power density might allow a better fuel economy, reduce refueling times, reduce personnel radiation exposures and waste handling and allow a better overall plant capacity, thus utilizing better the invested capital.

Favorable answers to the second and third questions strongly dictate that presently known and proven technology be used if the solution is to be regarded as a viable alternative by the electric power generating industry.

The Safer – Highly Available Reactor Plant (SHARP) concept described below appears to be a solution meeting these overall demands. It would definitely enhance safety by improving the core power distribution, thermal-hydraulics conditions, heat removal requirements and overall plant operations. It would not require significant new engineering and technical innovations, since minimal variations would be applied to the currently used reactor designs. These new plants will probably be easier to license than the current plants. The economic aspect is the only phase on which a conclusion is not easily forthcoming, and it requires in-depth analysis before reasonable cost comparisons can be established. The purpose of this work is to conduct an in-depth comparative and

optimization study of the fuel utilization capabilities of the Safer – Highly Available Reactor Plants versus the standard LWR plants currently in operation or under construction, and concurrently, to develop calculational tools that will facilitate and enhance the accuracy of scoping-type burnup studies. The primary study must be complemented with indications of the comparative safety and economic performance of the SHARP with respect to standard PWR plants.

1.2. The SHARP Concept

The main goals pursued by the "Safer – Highly Available Reactor Plant" (SHARP) are the following:

- Enhance reactor safety.

- Reduce the safety and operational problems caused by the operating conditions prevalent in current LWR systems.

- Reduce lengthy and costly research and licensing procedures.

- Offer a solution which utilizes current technology and can be implemented in a relatively short time.

- Reduce economic penalties to make a commercially viable operation.

The dominant feature of the SHARP is the use of a standard PWR core, without core changes requiring technological modifications, at a reduced power density to reduce safety-related and operational problems associated with standard full power operating conditions. The single low power density core may then be employed in an essentially standard plant configuration, yielding a reduced power

rating plant. Alternatively, it could be employed in parallel with several other low power density cores powering a single standard secondary side so as to yield a plant of normal or full power rating. This multi-core arrangement was first considered attractive to maintain the overall capital cost down and maintain large plant output. However, from the commercial point of view, it is obviously more reasonable to study the single-reactor reduced power rating plant, whose design and construction are well known. It is important to consider also that the economy of scale afforded by the present large plants lies in the reduced number of critical, expensive components used for the production of a large amount of power. The low power density concept breaks away from this constraint and attempts to demonstrate that there is no large advantage in designing large power output plants as compared to smaller sizes, when all factors influencing plant economics are considered. Furthermore, it is frequent to find electrical demand areas where the large power rating plants are oversized and a smaller plant is better suited for such places.

A single-core SHARP, which is the main concept analyzed in this work, is a plant using a standard, full-sized reactor vessel operating at a reduced power level, with the balance of plant dimensioned according to the desired rated output power (50% of standard, 33% of standard, etc.). The use of a single full-sized core for the low power density system accomplishes the dual goal of avoiding extensive plant redesign, and operating a core at a power level clearly within its technological limits with enhanced safety

and operational characteristics. A higher plant capital cost per unit of installed power is expected (1) because of the losses associated with the economics of size, i.e. because of the additional investment in the "oversize" Nuclear Steam Supply System (NSSS) components However, the overall cost of energy production would be competitive because of the improved operating conditions and safety.

Note that although the power density (and thus, the heat flux across the fuel elements of the core) is reduced, the thermodynamic conditions of the coolant would be maintained at the same level as in the standard plants to avoid loss of thermodynamic efficiency of the plant. This can be easily achieved by reducing the coolant flow across the core, within established heat transfer limitations. Reducing the heat flux in the fuel while maintaining standard coolant conditions reduces the temperature of the fuel, which improves fuel/clad thermal conditions, reduces stresses in the fuel, and hopefully reduces fuel pin failures.

The advantages that can be obtained from the SHARP are:

a). From the heat transfer point of view:

- Lower temperatures in the fuel pellets, due to the lower heat flux. This would imply reduced thermal-related damage to the pellets and to the cladding, and therefore, reduced pin failures.

- Reduced probability of reaching critical heat transfer conditions.

— Reduced heat stored in the fuel, and increased available heat capacity, i.e., in case of an accident the core is capable of retaining more heat before suffering damages.

— Milder accident conditions and emergency cooling requirements, due to the lower fuel temperature and lower power density (which means reduced decay heat generation).

b). From the neutronics point of view:

— Less Doppler broadening of resonances, and therefore extra reactivity available.

— Reduced xenon concentration due to the lower neutron flux, which means again some extra reactivity available.

— Reduced xenon oscillation problems due to the lower overall xenon concentration.

c). From the fuel cycle point of view:

— Longer burnup achievable from the same initial fuel, due to the extra reactivities mentioned above.

— Largely increased time between refuelings due to the double effect of larger burnups achievable and the lower power generation of each core. This would reduce the impact of refueling outages on plant availability tending to increase it. The increased availability would result in proportionally increased energy generation in a given time period, thus reducing the impact of capital (which would be a basically fixed total cost) on energy generation cost.

— Reduced relative activity of the fuel at discharge per MWd generated due to the largely increased residence of the

fuel in the core, which would allow more of the mid-life fission products to decay while in the reactor.

- Reduced ore requirements due to the larger burnups achievable from the same initial cores.

- Reduced enrichment needs.

d). From the operations point of view:

- Increased plant availability and capacity factor due to the longer inherent fuel cycle.

- Operations well within technological limits with reduced failure of components and reduced forced outage periods.

- Decreased operational transients and constraints.

- Better load following capability because of the large technological margins available for maneuvering.

- Reduced overall personnel radiation exposure.

- Reduced fuel handling cost because of less material being handled and because of its lower specific (per MWd generated) radioactivity (which results in reduced personnel radiation exposure).

- Possibility of reduced spent fuel storage and transportation requirements.

- Easier inventory control and reduced risk of proliferation because of the reduction of fuel handling operations.

All these advantages appear to be qualitatively obvious, but the question remains whether or not they can outweigh the economic disadvantage mentioned before.

The present study conducted on the low power density single-core, reduced power rating plant using essentially the standard plant configuration is also timely because of the present worldwide interest towards building new, small size plants. There are many situations where a 1000+ MWe plant is just too large, and so is the capital investment associated with it.

1.3. Previous Studies of Low Power Density Cores

The general trend followed by industry since the application of nuclear reactors for commercial production of energy has been to increase plant size and power densities in an effort to reduce the capital cost per KW installed. The trend of higher power densities reduced both the materials involved in the reactor construction and the fuel inventory necessary for a given plant power rating.

A few commercial reactor concepts having significantly lower power densities than the LWR's are in operation, but they present higher capital cost and reduced operating experience. Such is the case, for example, of the Magnox reactors.

Lower power densities for LWR's were, thus, not considered seriously until recently under the NASAP(2) project (Nonproliferation Alternative Systems Assessment Program) where low power density cores were investigated primarily as a means for increasing the fuel utilization in a once-through fuel management scheme. This would reduce fuel handling operations, and therefore proliferation risks, while also providing additional safety margins.

This low power density study for the NASAP project was carried out by Westinghouse Corp., and their approach to low power density was that of increasing somewhat the overall core dimensions, while maintaining or even augmenting the output power. This approach meant only a moderate reduction in power density (about 23% as compared to the 50 to 75% reductions considered in the SHARP concept). This system was found to be advantageous from the points of view of safety and fuel utilization when compared to the standard high power plants. On the other hand, the need for redesign of the core, the pressure vessel, fuel handling mechanisms, etc., was found to require large investments both in time and in capital, and the idea was not considered practical by the private industry unless governmental support was provided.

1.4. Purpose of this Study

The main purpose of this study is to research and develop the neutronic and fuel utilization characteristics of the SHARP concept, and for purposes of completeness, to assess its overall safety and economic capabilities in comparison with standard PWR plants.

Several steps are necessary in order to accomplish this task:

a). Definition of what constitues a SHARP for the purpose of this study (ranges of power densities, safety considerations and/or goals, etc.).

b). Definition of a set of possible reactor cores on which parametric studies can be effectively conducted.

c). Preliminary neutronic and fuel cycle study of the defined cores with known and reliable calculational tools in order to have a good estimate of the performance characteristics of each core. This would include obtaining first estimates of basic core parameters (such as reactivity worth of boron, coolant temperature, Doppler reactivity coefficient, etc.) and of the expected variation of core and fuel cycle parameters (burnup, cycle length, etc.).

d). Exploration and evaluation of currently available calculational tools that may be suitable for this type of study, and development of new calculational schemes that may accomplish the established goals with the optimum cost/results ratio and serve the nuclear industry as an accurate scoping tool.

For example, this study needs a good set of calculational tools for the neutronics calculations, where power levels, fuel temperatures, etc. can be easily specified and changed, since these are some of the main parameters that will differentiate the low-power core from the standard core. It also requires reliable and economical means of calculating core burnup distribution, core life time, etc.

e). Detailed neutronic and fuel cycle studies of the cores selected from the parametric variations performed in the preliminary study. This study must result in an optimization of the SHARP cores from fuel-cycle and plant operation points of view. The results of the SHARP study must be compared to those of standard PWR's analyzed with the same calculational tools, in order to obtain relative figures of merit with a minimum of

methodological errors that could bias the estimates towards either system.

f). Overview of economic evaluation of the SHARP concept as compared to the standard PWR to establish the overall economic advantages or disadvantages that may be expected from the SHARP concept. The economic evaluation must consider the variations in the costs of interest and escalation, as well as the possible size or rating variations of the plant components which depend on the plant power rating. For example, when studying a single-core SHARP for a power level of 50% that of a standard plant, the NSSS is dimensioned equal to that of the standard, full-power plant, but the BOP is dimensioned for only the new 50% power rating.

The next chapter describes the SHARP parameters and the preliminary calculations carried out in order to establish the basic expected performance of the low power cores.

Chapter III describes the burnup calculational methods used in the industry; they are compared to the needs of this study, and a new method is developed, which best suits the scope of this work and results in reduced computational effort. The new method should serve as a valuable industry-wide burnup calculational tool because of its accuracy, ease of utilization and low computational cost.

Chapter IV shows the cases chosen for in-depth study and the results obtained from the burnup calculations, plus some data referred to the expected comparative plant performance.

Chapter V is a brief description of the thermal-hydraulics safety-related aspects of the SHARP as compared to the standard plant, and an insight into the comparative economic behavior of the SHARP.

Chapter VI contains the main conclusions of this study and recommendations for future research on the topic, considering points that have appeared as unresolved and potentially advantageous questions concerning the SHARP concept.

CHAPTER II
SCOPING WORK

2.1. Problem Framing

The purpose of this chapter is to study a sequence of different low-power cores in order to calculate first estimates of the low-power cores characteristics. A reference standard core must be defined against which one may compare the characteristics of the low-power cores. Finally, the results of the comparisons of the different reduced power density cores versus the standard one will be used to define the characteristics of selected cores that will undergo an in-depth study. The possible need for new calculational tools to perform such in depth study will also be examined. This chapter will, therefore, study a series of low-power cores. The results of this study will be used to select the better candidate cores, and a range of their expected characteristics, for further analysis.

The study of the SHARP requires an initial definition of the parameters of the reactor and/or of the fuel cycle that may be used as a means of comparing the SHARP with the standard PWR plant. These parameters may be classified into three main categories of interest: Safety, Neutronics, and Fuel Cycle. These are key areas of investigation which have both independent and interrelated problems.

Parameters of interest from the safety point of view would be:

a). The moderator temperature reactivity coefficient (MTC).

b). The Doppler reactivity coefficient.

c). The fuel average temperature.

d). The power peaking factors.

e). The soluble boron reactivity worth (at the concentrations needed throughout a cycle life for each particular reactor system).

f). Power density and linear power (KW/ft).

From the neutronics point of view, some of the more important parameters to be observed are:

a). The changes of Doppler reactivity coefficient at the different power density levels.

b). Xenon concentration.

c). Core effective multiplication factor at beginning of life.

c). Evolution of isotopics.

d). Neutron energy spectrum variations.

e). Effects of varying enrichments if enrichments different from those normally used for present PWR's can be considered.

From the fuel cycle point of view, there are two main variables that have to be optimized with regard to energy production, but which are bound by conditions like the maximum power peaking factor, fuel enrichment costs, thermal-hydraulic safety aspects, etc. These two main variables are the following:

a). The total energy obtainable per ton of uranium ore.

b). The core cycle length.

Obviously, one is concerned with obtaining the maximum energy from each ton of ore. In addition to this, utilities are also interested in having long core cycles. Presently there is a trend to change from a 12-month to an 18-month cycle scheme because of the reduction in refueling outage time and personnel radiation exposure, which might result in attractive savings in power generation cost.

If the currently standard core structure were maintained (3-batch, out-in scheme) it would only take an increased fuel enrichment and some help from burnable poisons in order to have an 18-month cycle instead of a 12-month one. By so doing, the ore utilization is favored by the fact that the fuel is discharged at a higher burnup. A study by Combustion Engineering(3) shows that ore utilization is improved for increasing burnup levels up to about 50 GWd/MTU and fuel enrichments of about 4.5%. The problem, however, is that an 18-month cycle would yield high discharge burnup levels as compared to what is acceptable based on the present level of fuel technology.

In order to keep the discharge burnup in an acceptable range, given the present fuel technology, and to take advantage of the reduced refueling outages of an 18-month scheme, utilities are forced to switch to larger batch sizes. This enables the power generation cost to be reduced, but with worse ore economy than that obtained in a 12-month core cycle. This is of some concern from a fuel resources viewpoint, mainly if reprocessing is not considered.

The optimum fuel cycle appears to be one which obtains maximum energy per ton of ore and which also has long core cycles. Discharge burnups, however, must be limited, considering present fuel technology, to somewhere between 30 and at the most 40 GWd/MTU. In addition, the size of the batches should be kept small (i.e. the number of batches should not be reduced) in order to maintain good ore utilization.

The reduced power density cores are expected to yield increased burnups given a certain core management scheme, but they are not expected to vary in extremely large proportions. This would probably make the low-power cores able to comply with all the desirable conditions mentioned above: long core cycles because of the reduced power density and the associated additional burnup, and better ore utilization because of the extended burnup obtained from the same initial core load.

The comparison of the fuel cycle performance of the SHARP versus a standard PWR plant can be done from many different frames of reference, each of which would enhance the comparison of the cores in a particular aspect. However, the two most significant ones would probably be the following:

a). Maintaining feed enrichment and core structure, evaluate the differences in burnup achievable, cycle length, and ore utilization.

b). Maintaining core structure and discharge burnup, evaluate the differences in ore requirements, enrichment needs, and cycle lengths.

Another of the other possible scenarios for comparison could be to maintain the discharge burnup constant and allow fuel enrichment, cycle length and core structure to vary. However, it is the feeling of the author that these cases would not aid significantly in demonstrating the differences of a SHARP as compared to a standard plant.

Once the main parameters of interest have been defined, it is necessary to define a reference core that represents a standard reactor plant, and whose characteristics and performance is known and usable as a frame of reference for comparison of the SHARP characteristics and as a benchmark of the calculational methods used in the study.

After the reference core is defined, it is necessary to define a basic SHARP core and a series of "variational" cores that will be used in order to obtain the coefficients, parameters, and comparative results defined above. After the main SHARP characteristics are obtained, another set of cores will be selected for the in-depth study and comparison with the standard plant. The calculational tools needed for the in-depth study will be defined based on the requirements and restrictions observed in the scoping calculations.

The core chosen for the standard reference plant is a Westinghouse four-loop, 3400 MWth. PWR, with 17x17 pin fuel assemblies(4) whose main parameters appear on table 2.1.1. Table 2.1.2 describes the main characteristics of the basic fuel cell of this standard reactor.

Table 2.1.1. Main Core Parameters for the Standard Reactor.

Core Shape	Cylindrical
Radius	168.53 cm
Active Height	365.00 cm
Reflector Thickness	34.00 cm
Active Volume	3.2568 E+7 cm^3
Heavy Metal Loading	94.418 MTU
Array Geometry	Rectangular
Pitch	1.2573 cm
Coolant Pressure	2250 psia
Avg. Coolant Temperature	583 K
Thermal Power	3400 MWth
Pin Average Linear Power	6 KW/ft

Table 2.1.2. Basic Fuel Cell of the Standard Reactor.

Region	Material	Radius(cm)	Thick.(cm)	Vol. Fraction
1	Fuel	0.4096	0.4096	0.3334
2	Gap	0.4178	0.0082	0.0135
3	Clad	0.4750	0.0572	0.1015
4	Water	0.7094	0.2344	0.5516

The basic reference core used in the scoping study is described in Table 2.1.1; its fuel cell characteristics are described in Table 2.1.2. The fuel used for the reference core is uranium dioxide, enriched to 3% in U-235 isotope; the moderator is considered at standard operating pressure (2250 psia.), but at room temperature (293 K). No soluble or lumped poisons are considered, nor are any fission products present for all the scoping beginning of life (BOL) calculations, unless otherwise specified.

Under these conditions, the composition of the basic fuel cell is as shown in Table 2.1.3. Region 1 corresponds, as in Table 2.1.2, to the fuel pellet; Region 2 is the gap between pellet and clad; Region 3 is the Zircaloy-4 clad and Region 4 is the light water moderator-coolant. This reference core will be named core #1.

The pin-average linear power for the standard reactor is 6 KW/ft. A pin-average linear power of 1.5 KW/ft. is used for the scoping studies of the SHARP. This is one fourth the linear power of the standard Westinghouse core.

In an attempt to cover a wider scope and range of possible applications of low-power reactors, the scoping study includes some exploration of a very low power, low enrichment core which could be used as a preheater in a multi-core configuration similar to the one illustrated in Figure 1.2.2. This core would have lower than standard moderator temperature and a very low power (20% that of the standard core). Such a low power density, low moderator temperature core might possibly be fueled with spent fuel

Table 2.1.3. Basic Cell Composition. Core #1.

Isotope	Region	Pure # Dens.(*)	Cell Avg. # Dens.(*)
H	4	6.7 E-2	3.6957 E-2
O	4	3.35E-2	
O	1	4.4009 E-2	3.3152 E-2
U-235	1	6.6830 E-4	2.2281 E-4
U-238	1	2.1337 E-2	7.1137 E-3
Zr	3	4.2808 E-2	4.345 E-3
Ni	3	0... E-10	0... E-10
Sn	3	4.8556 E-4	4.9285 E-5
Fe	3	1.4946 E-4	1.517 E-5
Cr	3	7.6426 E-5	7.7573 E-6
He	2	1.9 E-3	2.565 E-5

* Units are atoms per barn-cm.

discharged from standard PWR plants. This core was thought of as a possibility for further use of standard plant spent fuel.

Table 2.1.4 identifies the cores used for the scoping study with their main distinguishing characteristics. The characteristics of these cores were selected in order to obtain indicative figures on reactivity coefficients and burnup variations. Each core has a case I.D. assigned to it, which is used for future reference. The power levels indicated are in percent relative to the standard core's full power (6 KW/ft. or 3400 MWth. total core power).

In all cases, both the fuel and the moderator are treated as having a uniform temperature throughout their masses. In the reduced power density systems, the moderator temperature is always kept at the average coolant temperature of the standard plant core, in order to maintain the thermodynamic characteristics of the steam cycle, and the thermal efficiency of the plant. The only cores with different coolant temperature are the ones intended for preheater operation. The fuel average temperatures depend obviously on the linear power density, and the values used for each of the scoping study cores are shown in next section.

Table 2.1.4 shows also the core effective neutron multiplication factor (K-eff.) obtained for each of the scoping study cores in a BOL, clean, unrodded situation. The calculational procedures used for the scoping study are different than the ones used for the detailed burnup study, and they are specified later in this chapter.

Table 2.1.4. Identification of Cases for the Scoping Study.

Case #	Enrich.	Mod.Temp.	Fuel Temp.	Poisons	Power	K-eff(*)
1	3%	293 K	293 K	0	0	1.3532
2	3%	583 K	293 K	0	0	1.2933
3	3%	583 K	1005 K	0	100%	1.2675
3A	2.6%	583 K	1005 K	0	100%	1.233
4	3%	583 K	700 K	0	25%	1.2770
4A	2.6%	583 K	700 K	0	25%	1.242
5	2.6%	293 K	293 K	0	0	1.3184
6	3%	293 K	293 K	1300 PPM	0	1.1579
7	3%	420 K	520 K	0	20%	1.3304
8	2%	420 K	520 K	0	20%	1.2260
9	3%	583 K	1005 K	1856 PPM	100%	1.0781

* Calculated with PHROG-BRT-MONA as shown in Sections 2.2.2 and 2.2.3.

Table 2.1.5. Tabulation of Case Number Densities Which Differ from Those of Case #1.

Case #	Isotope	Region	Pure #Dens(*)	Cell Hom. #Dens(*)
2	O	4	2.346 E-2	2.76149 E-2
2	H	4	4.6926 E-2	2.58845 E-2
3	Same as core #2			
4	Same as core #2			
5	U-235	1	5.7924 E-4	1.9312 E-4
5	U-238	1	2.1426 E-2	7.14335 E-3
6	B	4	7.258 E-5	4.0037 E-5
7	O	4	3.0982 E-2	3.17624 E-2
7	H	4	6.1964 E-2	3.41795 E-2
8	U-235	1	4.4561 E-4	1.48565 E-4
8	U-238	1	2.1559 E-2	7.18790 E-3
8	O	4	3.0982 E-2	3.17624 E-2
8	H	4	6.1964 E-2	3.41795 E-2
9	B	4	7.258 E-5	4.0037 E-5
9	O	4	2.346 E-2	2.76149 E-2
9	H	4	4.6926 E-2	2.58845 E-2
3A	U-235	1	5.7924 E-4	1.9312 E-4
3A	U-238	1	2.1426 E-2	7.14335 E-3
3A	O	4	2.346 E-2	2.76149 E-2
3A	H	4	4.6926 E-2	2.58845 E-2
4A	Same as core #3A			

* Units are atoms per barn-cm.

Some of the cores identified in Table 2.1.4 show variations in moderator temperature or fuel enrichment that imply changes in the cell number densities as compared to the reference cell. The coolant and fuel densities for the different temperature situations were calculated with the help of standard tables for each material(5,6). Table 2.1.5 shows the changed number densities for the affected isotopes and applicable cases. For the rest of the isotopes and/or cases, the reference core number densities apply.

2.2. Calculational Methods for the Scoping Study

2.2.1. Heat Transfer Calculations.

Some of the main differences between a standard reactor and a SHARP are related to the differences in the fuel temperature. It is therefore necessary to know the fuel temperatures that correspond to several different power-density cores before their neutronic study can be started. However, for the purpose of the present study, it is not necessary to obtain extremely accurate results, since the data will be used for obtaining varying nuclear properties of the fuel, which are not drastically affected by a few degrees of uncertainty in the temperature.

The calculational scheme used may be found in any complete Nuclear Heat Transfer or Reactor Analysis text(7) It is assumed that a uniform volumetric heat source exists in the fuel region; that the bulk moderator operating conditions are kept constant for all the different cores (583 K, 2250 psia.) except for the preheater cores (cases #7 and #8) which have their moderator at 420

deg. K but at the same 2250 psia pressure. For all the other cores, subcooled nucleate boiling has been assumed at the pin surface, while cores #7 and #8 were assumed to have subcooled convection, due to the much lower bulk coolant temperature and to the very low heat flux rate. Figure 2.2.1 shows the geometry assumed in the pin for the heat transfer calculations.

(Dimensions in cm)
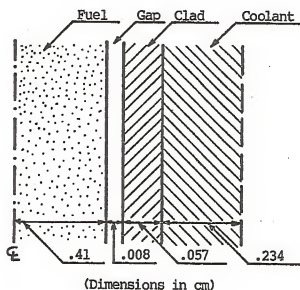
Figure 2.2.1. Pin Cell Geometry.

Given the heat transfer conditions found at the pin surface, the temperature increment between the bulk coolant and the clad surface may be obtained by the Jens & Lottes expression:

$$Ts = Tc + \frac{60 * (q'' / 10^6)^{0.25}}{exp (p / 900)}$$

where Ts = pin surface temperature, deg. F

Tc = bulk coolant temperature, deg. F

q'' = surface heat flow rate, Btu/hr sq.ft

p = coolant pressure, psia

The temperature increment across the clad is given by the expression:

$$\Delta Tcl = \frac{q' * do * \ln (do / di)}{2 * K * S}$$

where ΔTcl = increment of temperature across the clad, deg. F

q' = linear heat flow rate, Btu/hr ft

do = clad outer diameter, ft

di = clad inner diameter, ft

K = clad thermal conductivity, Btu/hr ft deg.F

S = clad surface per unit pin length, ft

Since the thickness of the gap is extremely small, the temperature change across the gap may be obtained as

$$\Delta Tg = \frac{q'}{K * S}$$

where ΔTg = increment of temperature across the gap, deg. F

q' = linear heat flow rate, Btu/hr ft

K = thermal gap conductance, Btu/hr sq.ft deg.F

S = surface of heat transfer per unit pin length, ft.

With the previous expressions, it is possible to obtain the pellet surface temperature. Since the temperature of interest for nuclear calculations is the average temperature of the pellet, it is necessary to know the temperature profile in the pellet, as a function of the distance to the centerline, so that volumetric weighting of the temperature can be done.

The temperature profile inside the pellet (assuming uniform heat generation) is given by

$$T(r) = To - \frac{q' * r^2}{\pi * R^2 * 4 * Kf}$$

where T(r) = temperature of the pellet at radius r , deg. F

    To   = temperature of the pellet centerline, deg. F

    q'   = linear heat flow rate, Btu/hr ft

    R   = pellet outer radius, ft

    Kf   = pellet thermal conductivity, Btu/hr ft deg. F

    r   = distance from centerline, ft

One more quantity is needed at this point: the pellet centerline temperature, which is given by

$$To = Ts + \frac{q'}{4 * \pi * Kf}$$

where To = pellet centerline temperature, deg. F

    Ts = pellet surface temperature, deg. F

With the pellet temperature profile expression in hand, the pellet average temperature may be obtained as

$$Tave = (2 \ / \ R^2) \ * \int_{0}^{R} T(r) \ * \ r \ * \ dr = To - (q' \ / \ 8 \ * \ \pi \ * \ Kf)$$

For the 20% linear power, preheater cores, the calculational scheme was the same except for the temperature step from coolant to clad surface, in which case a subcooled convection heat transfer expression was used, such as the following:

$$\Delta T = q'' \ / \ h$$

where q'' = surface heat flow rate, Btu/hr sq.ft

h   = forced convection coefficient, Btu/hr sq.ft deg. F

The forced convection coefficient may be calculated as

$$h = 0.0023 \ * \ Re^{0.8} \ * \ Pr^{0.4} \ * \ (K/D)$$

where K = coolant thermal conductivity, Btu/hr ft deg.F

D = hydraulic diameter of the coolant channel, ft

Re = Reynolds number

Pr = Prandtl number

Once the thermal calculations were completed, the standard core fuel average temperature was checked against the vendor's literature, and found to differ by a few degrees; this difference is probably due to slightly different conductivity coefficients across the core (due to temperature changes), which are not taken

Table 2.2.1. Basic Cell Temperatures vs. Power Level.

Power level %	100	50	33	25	20 (pre)
Clad surface	593.3	592.8	592.5	592.4	304.2
Clad increment	46.4	23.2	15.5	11.6	9.3
Gap increment	120.1	60.1	40.0	30.0	24.0
Fuel surface	759.9	676.1	648.1	634.0	337.5
Fuel centerline	1846.7	1219.5	1010.3	905.7	555.0
Fuel average	1303.3	947.8	829.2	769.9	446.3
Fuel average (K)	979.0	781.9	716.0	682.0	503.2
Vendor normalized (K)	1005.0	805.0	735.0	700.0	520.0

Note: Temperatures obtained from thermal calculations as described in Section 2.2.1.
    Temperatures expressed in degrees Fahrenheit, unless otherwise specified as Kelvin (K).
    Column 20 (pre) refers to the 20% power, preheater core.

into account in the thermal calculations. Since this difference was not significant from the nuclear point of view, all the results were modified slightly to maintain consistency with the vendor's data in the neutronic calculations. The consistent data permit the further benchmarking of the nuclear calculations. Table 2.2.1 shows the main temperatures involved in the calculation for all the different power levels studied. All temperatures are in degrees Fahrenheit, unless otherwise specified.

2.2.2. Neutron Cross Section Calculations.

Section 1 of this Chapter described the set of cores chosen for the scoping observation of the differences between a standard core and a low power density core. It is necessary to gather a series of codes of known reliability which can yield an accurate neutronic picture of a LWR core. Parametric studies can then be conducted which will define significant factors differentiating a SHARP from a standard plant. In these parametric studies, the absolute accuracy of the results is not as significant as their relative values which are used to establish figures of merit.

The best neutron cross section calculational method available for the scoping study involved the use of several standard codes. This method allows great flexibility in specifying isotopic number densities, geometries, etc., but it requires a relatively large amount of manual data handling from code to code. This section briefly describes these codes.

Two codes are used for the calculation of cross sections. PHROG(8) is used for calculation of the fast-group neutron cross sections, and the original library, consisting of a 68 energy group cross section data set is collapsed into three fast groups which are used in the core model code. A more detailed explanation of PHROG can be found in Appendix B2. As is shown in the next section, the core criticality calculations are done with a model which defines two distinct regions: fuel and reflector. Fast cross section calculations need to be run for both of these regions. The fuel region is run first, and the group-wise fluxes and currents existing in the fuel are used as weighting functions for the calculation of the reflector constants.

The thermal-group cross sections are calculated with the Battelle-Revised Thermos (BRT) code(9) using a 30-group cross section library. Thermal cross sections are collapsed into a single thermal group. It is also necessary to obtain separate cross sections for the fuel region and for the reflector. The fuel region does not present any problem, since it can be well represented by the calculation of a unit fuel cell. However, BRT does not allow an intrinsic representation of the reflector region without an adjacent core. The reflector region is then calculated from a two-region slab reactor configuration, where one of the regions has the average core region characteristics, while the other represents the reflector. The geometry chosen was that of a slab. Because of limitations in the number of mesh spaces allowed by BRT, inaccuracies at the core-reflector boundary arise when a

cylindrical shape is considered. The basic features of BRT are described in Appendix B1.

Four sets of cross sections are obtained with the calculations described above: one set with three-group fast cross sections for the core region; another set with three-group fast cross sections for the reflector; a third set with thermal cross sections for the core, and finally a set with thermal cross sections for a slab core-reflector configuration, from which only the reflector constants are used. These cross section sets are then organized for input to the core models, which are discussed in the next section.

2.2.3. Criticality and Burnup Methods.

The first objective of the scoping study is to obtain the neutron multiplication factors of the cores described in Table 2.1.4. These multiplication factors are used to estimate the main reactivity coefficients involved in the neutronic aspects that differentiate the SHARP core from the standard core. The second goal of the scoping study is to obtain a first estimate of the burnup levels achievable by each particular core, as well as the isotopics associated with them. Following is the description of the methods used for achieving both of these purposes.

The core modeling for the criticality calculations is done with the MONA(10) code which is described in more detail in Appendix B3. MONA accepts the cross section input prepared by PHROG and BRT, and information about the core geometry and material region compositions. The code can perform a number of

calculations, such as buckling searches, poison searches, etc. In this case, a normal effective multiplication factor (K-eff.) calculation is requested.

The code accepts a cylindrical geometry, but it is a one-dimensional diffusion-theory code. In order to obtain accurate results for the neutron multiplication factor in one-dimensional cylindrical geometry, accurate values for the perpendicular bucklings to account for perpendicular leakage are required. In the absence of this information, it is known from previous experience that an "equivalent" spherical system can yield accurate results for the neutron multiplication factor. For the "equivalent" spherical configuration, the core volume is kept equal to that of the cylindrical reactor, while for the reflector, the thickness is kept equal to the cylindrical reactor value. The core region has 199 mesh points allocated, and 34 mesh points are assigned to the reflector, thus assuring that the mesh spacing is smaller than the neutron diffusion length. This is necessary for these calculations if neutron diffusion theory is expected to describe the core fluxes and currents with an acceptable level of accuracy. An extrapolated zero-flux boundary condition is specified for the outer boundary. Figure 2.2.2 illustrates the geometric model used with MONA for the criticality calculations.

Two methods were used for the scoping burnup calculations. The first method involves the use of the calculational scheme already used for the criticality calculations. Once the reactor criticality status is established, soluble boron is added to the
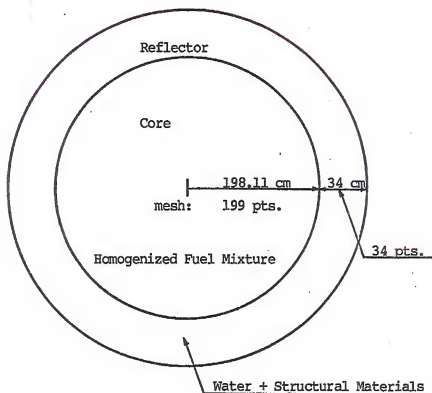
Figure 2.2.2. Geometry Used for MONA Criticality Calculations.

moderator in the amount necessary to force the core to be exactly critical.

With the reactor critical, the fuel-region homogenized composition that was input to MONA and the flux spectrum generated by MONA for the fuel region are input into a BURNUP code, which performs the isotopic burning of the mixture to the burnup degree specified, and at the power level desired. The BURNUP code used in this case is geometrically non-dimensional, but it allows for the specification of a number of time-steps after which the homogenized composition of the fuel region is recalculated. The flux spectrum is assumed to remain constant throughout the burnup calculation.

After the burnup calculation, the homogenized fuel-region composition is used again for a new core criticality evaluation, in the way explained above. Figure 2.2.3 shows a schematic diagram of the flow of data involved in this type of burnup calculation.

This burnup calculation scheme has some very obvious drawbacks:

a). Each burnup timestep requires an inordinate amount of effort.

b). There is a large amount of punched card handling and typing, which largely increases the probability of human error in the calculations.

c). The BURNUP code used has a rather low degree of sophistication, resulting in a reduced number and complexity of radioactive chains.
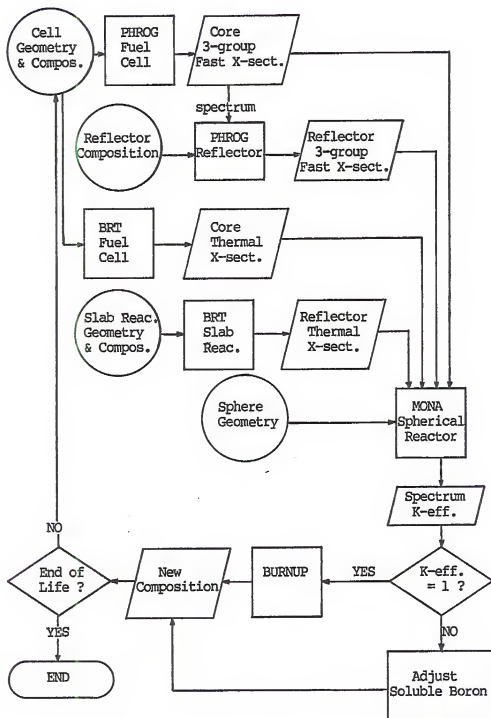
Figure 2.2.3. Code and Data Flow for Preliminary Burnup Calculation.

This method was, however, used for some cases, but a faster and at least equally accurate method had to be obtained. The second burnup method employed the LEOPARD program(11). This program, which is discussed in more detail in Appendix B4, makes an automated chain of calculations involving cross sections, spectra and burnup for an elementary fuel cell. The cross sections calculations are based on the MUFT(12) and SOFOCATE(13) codes. The geometry is always that of a fuel pin, which may be surrounded by a buffer zone simulating the structural materials present in the core but not associated with the elementary fuel cell.

LEOPARD requires an initial pin composition, a power level, a soluble boron concentration history, and a burnup timestep structure. The code calculates each isotope's cross sections collapsed to a specified number of broad groups, performs a multiplication factor calculation, burns the fuel, recalculates the pin composition, and resumes the loop until the completion of all the burnup timesteps.

Although the cross section generation in LEOPARD is not quite as accurate as the one used for the criticality calculations, the burnup calculational structure is far superior to the one in the BURNUP code, and the automatic coupling of the calculations allows for the effortless and error free handling of a much larger number of isotopes, which definitely adds to the accuracy of the overall calculation. It is important to note the drastic reduction of effort required for similar results between the first (PHROG-BRT-MONA-BURNUP) and the second (LEOPARD) method of burnup

calculation. Obviously, LEOPARD does not model the core and the reflector since it is just a pin cell, one-dimensional code, but neutron leakage effects are considered by means of a perpendicular buckling, which enables the code to perform rough estimates of criticality.

In an effort to compare the two burnup calculational methods, parallel calculations were run for one core representing a standard reactor's first load. Since no fuel regions were considered in the MONA core model, the fuel enrichment was specified as an average of the concentrations of the three real batches. This was also the only way of simulating the situation in LEOPARD, because it obviously can not accept several enrichments simultaneously. Note, however, that this is a rather crude way of representing a core, because it is very different to have three distinct regions with various enrichments or burnup levels than to have a large region with averaged characteristics. However, the procedure was deemed adequate for a scoping comparative study.

The results from the LEOPARD and MONA-et-seq. comparison showed a discharge burnup discrepancy of about 13%. Figure 2.2.4 shows the estimated soluble boron letdown curve obtained from each calculational scheme. The agreement is very good up to about two thirds of the core life, but the separation increases thereafter. This is most probably due to the absence of proper treatment of the fission products in the MONA method, because they could not be included in the fuel pellet region, and therefore were only accounted for as a reduction of core reactivity, but their
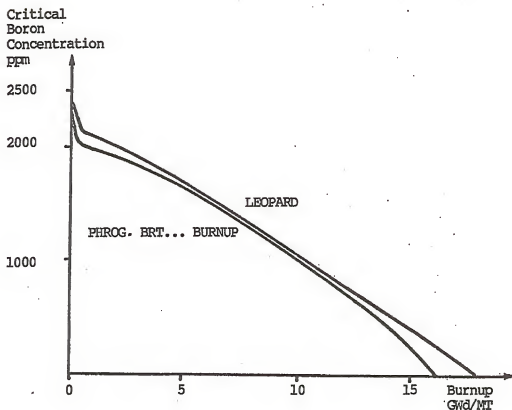
Figure 2.2.4. Comparison of Boron Letdown from LEOPARD and BURNUP.

neutronic effects on the other isotopes (due to spectral effects) could not be simulated. However, the agreement between the two methods can still be considered good, considering the different conceptual approaches, and the small amount of core information included in them. The most reassuring fact is that the quoted beginning of life (BOL) core reactivity and the quoted end of life (EOL) burnup level of the first core for the standard reactor lie just between the values obtained from the two models, with LEOPARD giving a better result for the EOL burnup.

With all these considerations in mind, LEOPARD was used with the "average fuel" enrichment for the scoping burnup calculations. These calculations used a fuel pin with core-averaged enrichment, a perpendicular buckling corresponding to the axial leakage of a standard core, and an approximately critical boron letdown curve. Calculations were performed for the standard core, a 25% power density core using varying lattice pitches, and a 20% power density core with reduced moderator temperature (the preheater core, designated as core #8 in Table 2.1.4). The results of the scoping calculations are shown in the next Section.

2.3. Results of Preliminary Work

2.3.1. Reactivity Coefficients.

Table 2.1.4 shows the main parameters characterizing the different cores that were chosen for the scoping study of the SHARP. The table includes the effective multiplication factors (K-eff.) of the cores at beginning of life (B.O.L.) with no xenon

or control rods, calculated by the PHROG-BRT-MONA scheme, as explained in sections 2.2.2 and 2.2.3.

The core effective multiplication factors obtained for the different cores are used to calculate bulk reactivity coefficients. Thus, the average moderator temperature reactivity coefficient (M.T.C.) is -20.67 p.c.m./deg.K of moderator temperature variation for the temperature range between 293 K and 583 K. This rather large reactivity coefficient would be the main asset of a core such as core #8, which would use very low enriched fuel or even spent fuel discharged from a standard plant. A reduction of average moderator temperature of 163 K, as described on table 2.1.4, would provide 3.37 % of extra reactivity as compared to an identical core working with normal moderator temperature. This excess reactivity would enable the core to work with the less reactive fuel or burn the standard fuel beyond the current discharge burnup levels. However, it should be kept in mind that these preheater cores are in some way a technical speculation, and that the real interest of this study is centered in the low-power cores with standard moderator conditions, which enables them to drive a standard balance of plant in a single-core arrangement.

As the average linear power produced by the fuel is varied according to the different power densities of the SHARP cores, so is the average fuel temperature, even though the moderator conditions remain unchanged. This variation of fuel temperature causes a change of the absorption resonances width, due to the Doppler effect, and this causes a change of the core reactivity.

The Doppler coefficient of reactivity obtained from Table 2.1.4 is −3.99 p.c.m./deg.K of fuel temperature change, for a variation between 293 K and 700 K (which correspond to 0 power and 25% power respectively). The coefficient drops to −3.12 p.c.m./deg.K for the temperature range between 700 K and 1005 K (25% power to full power range). This shows the well known behavior of saturation of the Doppler effect for increasing temperatures of the fuel. The average coefficient that applies for the total range from 0 power to full power is −3.62 p.c.m./deg.K . The reduced fuel temperature is one of the main effects contributing to an increase in the reactivity of a low-power core with respect to the standard core. As was mentioned before, the moderator conditions would remain unchanged for the different power level cores for thermodynamic reasons, and only the fuel temperature would experience a variation due to the change of the heat production rate.

Fuel enrichment obviously has an effect on core reactivity, and therefore, it is also possible to define a reactivity coefficient for it. For enrichments between 2.6% and 3% in U-235, the reactivity coefficient oscillates between 8.705% / 1% enrichment variation for a cold, clean reactor and 8.643% / 1% enrichment variation for a hot, clean, full power reactor. For the enrichment step between 2% and 3% of U-235, the 1/5 power, preheater core shows a reactivity coefficient of 10.44% / 1% enrichment variation. This larger magnitude may be due to the double effect of bracketing a lower enrichment end, and having a better moderated configuration due to the low moderator temperature

which implies higher moderator density. This rather large reactivity coefficient associated with the fuel enrichment could be used in order to stretch burnup of the cores by increasing the fuel enrichment, which up to a certain point will result in a better ore utilization (14).

Higher fuel enrichment would, however, bring problems associated with higher power peaking requiring the permanent use of burnable poisons and the technological problem of loss of clad integrity at very high burnup levels.

The reactivity coefficient of soluble boron in the moderator is given an in-depth study for two main reasons: First, the soluble boron concentration that makes a reactor critical at the different points through the core's life is a very useful and realistic way of estimating the cycle life of the core. However, for this to be feasible, it is important to know the reactivity worth of soluble boron at all times through the core's life. This coefficient may be expected to vary, depending mainly on boron concentration and core burnup status; these two variables are somewhat related, but they certainly don't follow any exact relationship. Second, just reducing the power density may cause the standard reactor lattice geometry not to be the optimal one from the burnup point of view, since the nuclear characteristics of the fuel are somewhat perturbed. Some changes in lattice geometry, such as fuel pitch may be required then in order to optimize the core for the new low-power situation. If this is the case, the ratio of moderator-to-fuel is likely to be changed, and then, a given

variation of soluble boron concentration would mean different total amounts of boron being added to or retrieved from the core, which implies that the soluble boron reactivity coefficient may also change for different lattice geometries.

Soluble boron worth variations are analyzed as a function of boron concentration, core burnup, core power level, moderator density, and basic cell geometry. These calculations were based on the LEOPARD scoping core burnup calculations.

When adjusting soluble boron concentration to keep the reactor critical through life, the worth of boron decreases slightly from the B.O.L. until about 2000 or 3000 MWd/MTU and increases thereafter, with a very slowly increasing slope. This variation is shown in Figure 2.3.1. This figure shows also that the variation of power level, without variation of moderator conditions, causes practically no change in the boron reactivity coefficient. On the other hand, either a variation of pitch or a change of moderator density (temperature) cause sharp changes in boron reactivity worth. In both cases, as expected, boron reactivity worth increases for the changes that imply a larger absolute amount of boron present in the core (i.e. for an increased pitch or for an increased moderator density).

Figure 2.3.2 shows more clearly the effect of lattice pitch on boron worth. It is also more evident that the variation of boron worth at critical through life is larger for larger pitches. The cause of the monotonical increase of boron worth with life is due to two adding effects: the self-shielding effect of boron itself
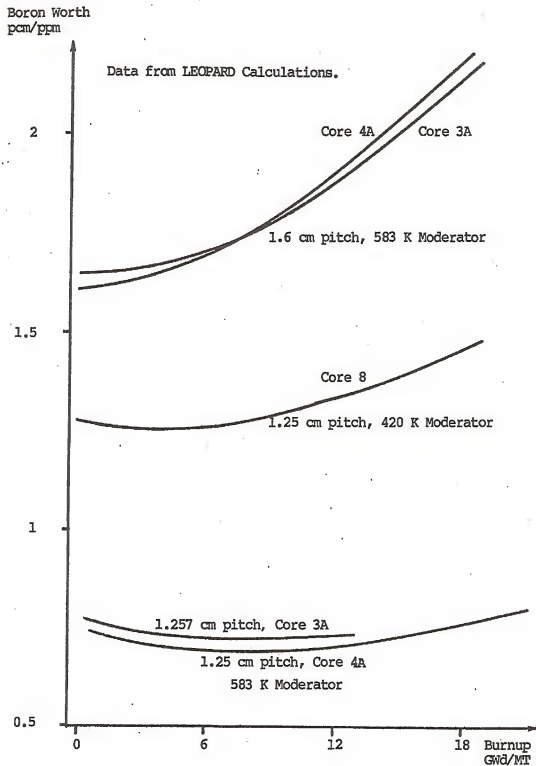
Figure 2.3.1. Soluble Boron Worth at Critical vs. Burnup.
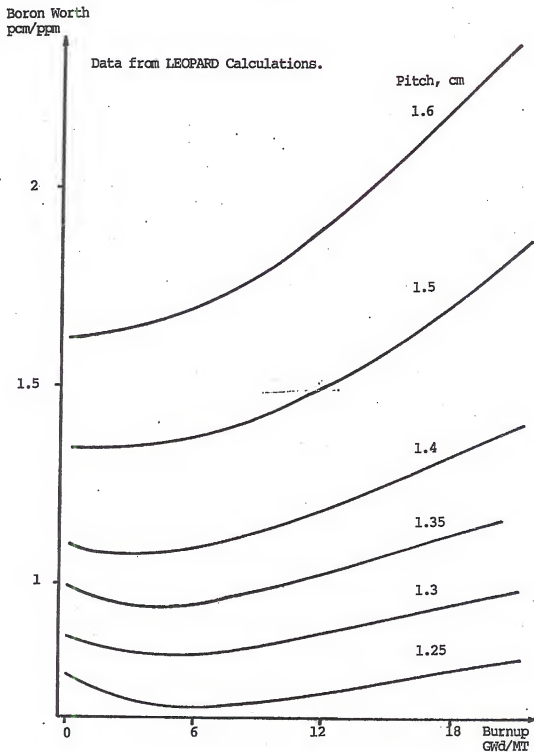
Figure 2.3.2. Soluble Boron Worth at Critical. Changing Pitch.

and the buildup of Pu-239. As core life advances, the boron concentration is steadily reduced; the boron self-shielding also decreases, causing an increase in boron worth. The contribution of Pu-239 buildup to the increase of the boron worth increases with core life. It should be noted that by the end of the core life, about 40 % of the core power is generated by Pu-239 that has been bred through parasitic captures in U-238. Pu-239 has a higher thermal fission cross section than U-235, and it has a resonance peak in the thermal energy region (tenths of an eV.). This causes the reactor to increase its sensitivity to the presence of a thermal poison (such as boron) that can remove neutrons from the high fission cross section energy region.

The rate of increase of boron worth with core life is faster for larger lattices, because of the effect of increased boron inventory in larger lattices.

Thus, from the point of view of the SHARP, it is possible to conclude that the reduced power density will not affect boron worth to a significant extent as long as the moderator conditions and cell geometry characteristics remain unchanged. However, if geometry changes are performed in order to optimize the cell behavior for burnup or fuel utilization, boron worth may be expected to show a significant variation, tending to increase for larger pitches and tending to increase towards E.O.L., mainly if the burnup levels achieved are high.

An important goal of the scoping study, is to make a first estimate of the burnup advantages that can be expected from the low

power cores relative to the standard core. One simple way of making such estimates is to calculate the total reactivity of the cores at B.O.L., and to assume that reactivity is lost linearly as core burnup increases, which is a reasonable approximation for a first-estimate calculation.

The most significant parameters altering the core reactivity at B.O.L. when the fuel enrichment, core geometry and moderator conditions are fixed, are the fuel temperature and the equilibrium xenon concentration. Fuel temperature affects the core reactivity in the amounts determined by the Doppler coefficient, explained earlier in this section. The reactivity worth of xenon in the different cores can not be obtained from the LEOPARD burnup calculations performed in the scoping study, but it can be obtained from the vendor's literature (4). Combining these xenon reactivity modifications with the core effective multiplication factors listed in Table 2.1.4, it is possible to obtain a comparison of the reactivity available at B.O.L. for each core, and therefore, an expectation of their comparative discharge burnup levels. These figures appear on Table 2.3.1 for the standard core, the 25% power level core, and the 20% power level preheater core. Two fuel enrichments are considered for each power level.

Using the linear reactivity assumption (15), Table 2.3.1 shows that a 25% power SHARP may yield a discharge burnup 10.6% above that of a standard reactor, for an average fuel enrichment of 2.6% U-235 in both cores. The burnup increase of 9% appears for a fuel enrichment of 3.1%. These burnup increases are desirable from two

main points of view: they represent additional energy obtained from the same initial ore, and they entail longer core cycles, which tends to reduce costs of refueling outages.

The expected burnups shown by the 20% power, preheater core are really surprising. The better moderation boosts reactivity in such a way that for 3% enriched fuel, a burnup increase of 32% above that of the standard core would be expected. If the preheater core is loaded with low enriched fuel, e.g. an enrichment of 2%, it would still yield 88% of the burnup expected from a standard reactor loaded with 3% enriched fuel.

However, it should be kept in mind that the burnup results obtained from these LEOPARD calculations do not take into account the significantly different neutronic situation of the various batches present in the core, which will definitely affect the isotopic dynamics, and therefore the core life. In any case, these figures give good hope for better fuel utilization by the SHARP's.

Table 2.3.1. Core Reactivities at B.O.L.

Case #	ENRICH	K-eff.	Xe worth	BOL reactivity
3	3%	1.268	-3.5%	23.3%
3A	2.6%	1.233	-3.5%	19.8%
4	3%	1.277	-2.3%	25.4%
4A	2.6%	1.242	-2.3%	21.9%
7	3%	1.330	-2.1%	30.9%
8	2%	1.226	-2.1%	20.5%

2.3.2. Neutronics and Safety Considerations.

The reduction of power density in the cores of the SHARP causes, among other effects, a reduction of the Doppler effect, a reduction of xenon concentration, and a reduction of fuel temperature. These changes entail unavoidable differences in the neutronic characteristics of the SHARP as compared to the standard reactor. Thus, it is necessary to check the neutronic changes associated with the power reduction, because some of their aspects are closely related to plant safety, while others are related to fuel performance or isotopic evolution.

As was pointed out in the preceding section, the neutronic differences existing between the standard reactor and the low-power reactors may result in a difference in the optimum fuel arrangement in the core. In this study, the variation of fuel arrangement is limited to the variation of fuel lattice pitch. This variation implies a change of the thermal-hydraulic characteristics of the fuel cell, as well as a change of the fuel-to-moderator ratio, whose heavy effect on the core neutronics is well known. The pin geometry is kept constant and equal to the standard core design.

The immediate effect of changing the pitch or fuel-to-moderator ratio is a change in the neutron energy spectrum, which in turn causes a change of cell reactivity, and as fuel burnup progresses, it affects the isotopics evolution. There is an optimum pitch which yields the maximum reactivity for a given fuel pin. However, such a configuration is not safe from the operational point of view. At the optimum point or in an

over-moderated configuration, the moderator temperature coefficient (MTC) becomes zero or positive, making the core inherently unstable. Stability and safety require a negative MTC, which means a pitch must be selected which yields an undermoderated configuration.

The scoping study included an exploration of the reactivity changes occurring in the fuel cell as pitch is varied, for some of the more representative core cases being considered. Figure 2.3.3 shows the plots of the infinite multiplication factor versus pitch for the full power standard core (core 3A), the 25% power core (core 4A) and the 20% power core with lower moderator temperature (core 8) as obtained from PHROG-BRT-MONA. The plots correspond to BOL, with no poisons, control rods or xenon present in the core, but with the temperatures corresponding to the assigned core power levels.

The curves show a uniform increase of reactivity between core 3A and core 4A, due to the reduced Doppler effect. Remember that no xenon effect is taken into account in these curves, since the cores are clean and with zero burnup. Curves 3A and 4A do not show, however, any visible change of shape; there is only a vertical shift due to the gained reactivity. This means that the best moderated pitch is the same independently of the power density (under equal moderator conditions). With this in mind, the overmoderation / undermoderation safety criteria for the low-power cores would admit some relaxation as compared to the standard core, due to the inherently safer operating conditions of the SHARP.
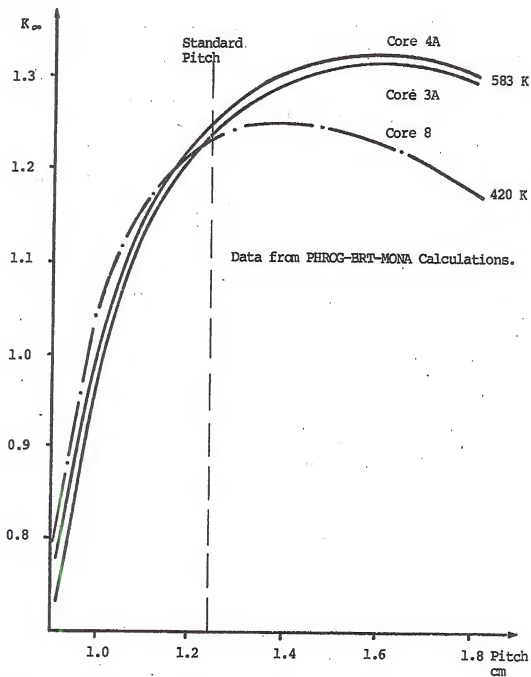
Figure 2.3.3. Infinite Multiplication Factor vs. Pitch.

Figure 2.3.3 shows the pitch used by the standard reactor. Note that it complies with the safety criterion of working in an undermoderated configuration. However, it seems that the pitch could still be allowed to increase somewhat without compromising the safety, since the operation point of the standard reactor is rather far from the peak of the curve. Later in this section it is explained why the standard reactor can not take advantage of the extra reactivity achievable from a slight increase of pitch, while the low-power core can.

When examining the curve for core #8 on Figure 2.3.3, a sharp change of shape can be noticed as compared to the other curves. Reactor 8's curve has a larger curvature; its reactivity is much more sensitive to the variations of pitch. This fact is easily explainable: Reactor 8 works with reduced moderator temperature, but at the same pressure as reactors 3A and 4A. The moderator has, therefore, a higher density. This can be observed on Table 2.1.5. The higher moderator density causes a given change of pitch to represent a larger increase in moderation, i.e. there are more mean free paths or more interactions taking place in the moderator region between pins at higher moderator density. It is as if the x-axis of the curve for core #8 has been compressed as compared to the standard moderator temperature ones.

One of the consequences of the higher moderator density is a reduction of the distance between the standard reactor working pitch and the optimum moderation point. This effectively raises the multiplication factor of core 8 when using the standard pitch,

and it is one of the main factors responsible for the high reactivity found for core 8 in spite of its low fuel enrichment (see Table 2.3.1). However, it is also important to note that this same curve shift prevents considering a different pitch for core 8, because it is already working at a point much closer to the optimum moderation ratio, and any further change of pitch would bring the core too close to a possibly overmoderated configuration, which as explained above, is not acceptable from a safety point of view.

Figures 2.3.4 and 2.3.5 show the changes in the four factors of the "four-factor formula" for cores 3A, 4A and 8, as pitches are varied. These are the results of the criticality calculations performed for BOL with PHROG-BRT-MONA. It is apparent that the neutronic effect of the higher moderator density of core 8 is far larger than the change caused by the Doppler effect difference between cores 3A and 4A. None of the four factors exhibit a difference of more than one percent in their corresponding values between cores 3A and 4A. However, the largest differences are observed in the resonance escape probability for very undermoderated lattices. This could be expected because this is the configuration and the factor that give the maximum enhancement to the Doppler effect variation.

The change of moderator density causes a sharp difference of spectra which is reflected in all the four factors. The fast fission factor, epsilon, is consistently lower for core 8 than for the other two. Note that the fast fission factor experiences a sharp rise when the lattice pitch reaches very small values and
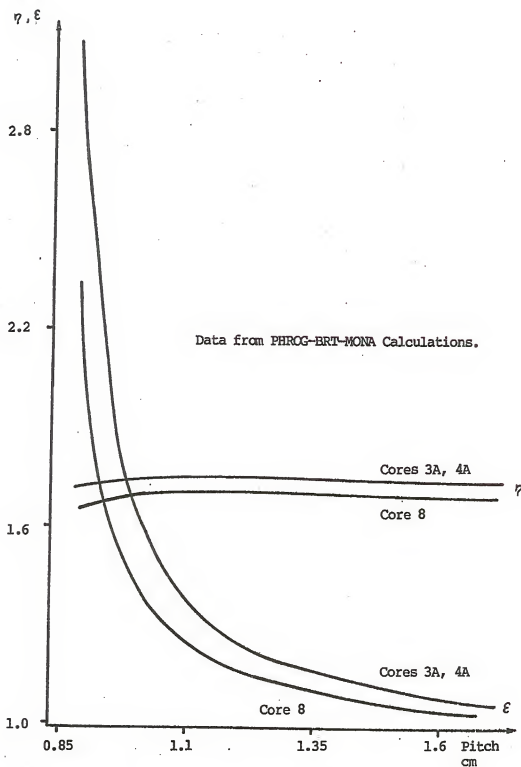
Figure 2.3.4. Eta and Fast Fission Factor vs. Pitch.

there is almost no moderator in the cell. In the range of pitches that can be reasonably considered for practical purposes (between 1.2 and 1.4 cm) the fast fission factor slowly decreases as more moderator is brought into the cell.

Eta, the number of neutrons produced per thermal neutron absorbed in the fuel, is the least sensitive of the factors, as it remains almost flat through the whole range of lattice pitches; it only shows a fall for very tight configurations. However, it is important to note that there is still a difference between core 8 and cores 3A and 4A. This shows that more important than the amount of moderator present in the cell is the variation of mean free path associated with the change of moderator density.

The resonance escape probability , p, shows a very predictable pattern, increasing as the amount of moderator increases, but showing a very clear saturation effect as large pitches are reached. However, in the range of practically reasonable pitches, p shows a rather steep positive slope; this factor is the one which is primarily responsible for the increase of reactivity achieved by increases of pitch in this range. Note that core 8 stays consistently above cores 3A and 4A throughout the range of pitches examined in spite of the larger relative number of resonance absorber atoms (mainly U-238) present in core 8. This is a direct effect of the different mean free path caused by the increased moderator density. If only the better moderation effect and saturation effect of p for larger pitches existed, the distance between the curves for core 8 and cores 3A and 4A could be expected
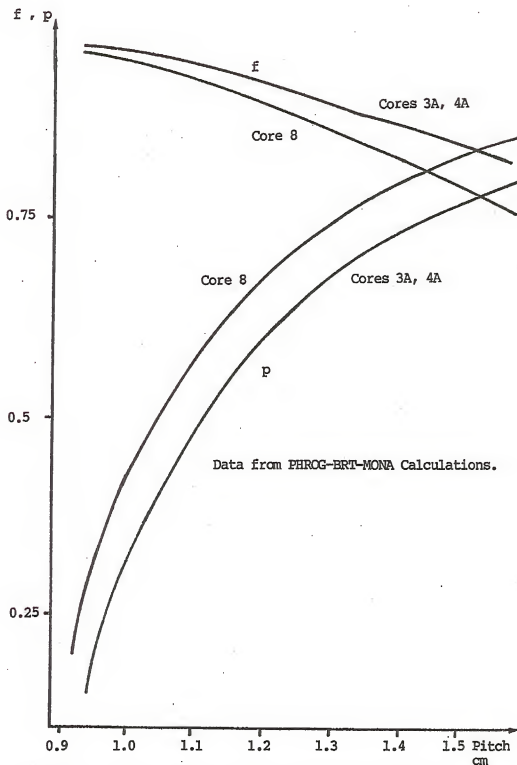
Figure 2.3.5. Resonance Escape Probability and Thermal Utilization.

to decrease significantly as larger pitches are reached, but the reduction of distance is insignificant, indicating again the key role of the changed mean free path.

Finally, the thermal utilization factor, f, or ratio of thermal neutrons absorbed in the fuel to total thermal absorptions in the cell, shows a drop as pitch is increased. This effect is directly related to the amount of absorbing moderator atoms present in the fuel cell. Note that as pitch increases, the curve for core 8, which is naturally below that of cores 3A and 4A, drops faster than that of cores 3A and 4A; the reason being simply that the same increase of pitch adds more moderator atoms into core 8 than for the other cores, because of the higher moderator density of core 8.

Many of the reactivity and isotopic differences between the standard core and the low power density cores have been attributed to neutron energy spectral effects. Figure 2.3.6 illustrates some aspects of the spectral changes. The graph represents the ratio of thermal flux to first fast group (of the three fast groups used for the criticality calculations) for cores 3A, 4A and 8, and for short burnup times, up to about 50 days of equivalent full power operation of the standard reactor. These figures were obtained from short burnup calculations performed with the PHROG-BRT-MONA-BURNUP scheme. The effect of the increased moderator density of core 8 is much larger than all the other effects differentiating cores 3A and 4A. However, all cores show a parallel behavior as burnup increases: there is an immediate drop of thermal flux right at the beginning of life. This may be
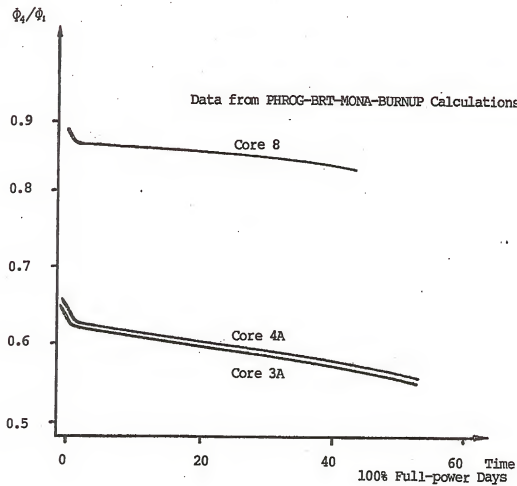
Figure 2.3.6. Thermal-to-Fast Flux Ratio.

attributed to the appearance of xenon. Note that the cores are always burned in the critical condition, which means that at BOL there is a high concentration of soluble boron keeping the core at critical. This boron concentration is rapidly reduced as xenon builds up to an equilibrium level. It would seem then, that the core is changing one thermal poison for another, and therefore no significant change of neutron spectrum would occur, but this is not the case. The fact that the thermal poison is in the fuel (in the case of xenon) as opposed to being in the moderator (in the case of boron) has a significant effect on the neutron spectrum. This is due to the different relative changes of thermal absorption cross section occurring in the fuel and in the moderator. Thus, as burnup increases, the thermal fraction of the neutron spectrum continues to drop slowly (the spectrum hardens) due to the generation of fission products, which are thermal poisons located within the fuel, as is the case with xenon.

A key point in the safety of operation of a new reactor is its thermal-hydraulic behavior. It has been stated before that the low power reactors should deliver the outgoing coolant in the same thermodynamic conditions as the standard reactor, because this is a basic factor in the thermal efficiency of the plant. It is obvious that if a reactor delivers a reduced amount of power, it is also forced to deliver a reduced coolant mass flow in order to keep the thermodynamic conditions at the outlet unchanged, and this flow reduction encompasses different characteristics in the heat transfer conditions.

One of the main parameters of concern when analyzing the heat transfer conditions in a PWR is the minimum DNBR (Departure from Nucleate Boiling Ratio) existing in the core. The minimum DNBR indicates how far the core is from reaching a film boiling heat transfer condition, which would result in heavy overheating and damage of the fuel.

Mark Miller performed calculations(16) on the subject of the DNBR for several power densities and several flow rates and compared the results to the DNBR actually found in the standard reference core. A more detailed analysis of the thermal-hydraulic performance of the low-power cores is presented in Chapter V. It is, however, important to point out here that all the reduced-power cores show a higher DNBR than the standard core, under equal outlet enthalpy condition, and using the same fuel lattice geometry. For larger pitches (which may be of interest from the discharge burnup viewpoint) the low-power cores can accept some pitch increase and maintain the outlet enthalpy and still stay at a better DNBR than the standard reactor. However, the standard core can not use larger pitches without seriously reducing its thermal-hydraulic safety margins, due to the reduction of coolant velocity associated with the increase of fuel lattice pitch. This is an unequivocal and key point proving the substantially higher safety level of the low power reactors as compared to the standard ones.

2.3.3. Burnup Achievements and Isotopic Inventories.

As explained in Section 2.2.3, the preliminary burnup calculations were performed with the LEOPARD code, using a pin enrichment equal to the average enrichment of the core, and burning it with a varying soluble boron concentration that would keep the core as close as possible to criticality. Burnup explorations were done for the standard reference core 3A, for the 25% power density core 4A, and as a scope extension, for the 20% power density, preheater core 8. All of them were examined at the standard core's pitch of 1.2573 cm; cores 3A and 8 were also investigated at one larger pitch, and core 4A was studied for a wide range of pitches, since it is the most representative low power density core. The multiple burnup calculations performed for core 4A have the purpose of determining the parametric effect of pitch on the discharge burnup level, as well as giving an indication of the optimum pitch value.

Table 2.3.2 shows the main burnup and isotopic results obtained from the preliminary burnup calculations.

Reducing power density to 25% of the standard level while keeping the standard core's pitch results in an increment of burnup of 9%. However, the burnup increment can be improved to about 16% by increasing the pitch of the low power reactor to about 1.3 or 1.4 cm. Figure 2.3.7 shows the burnup levels achievable as pitch is varied on core 4A. Note the important fact that the maximum burnup is not achieved at the pitch value having maximum BOL reactivity (shown in Figure 2.3.3) but at a smaller pitch. This
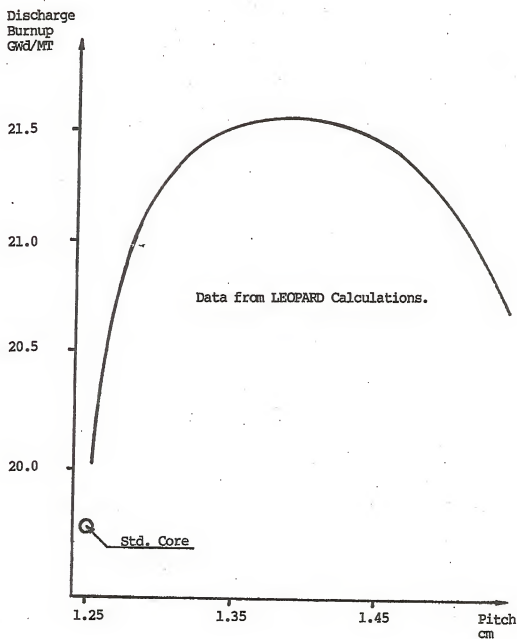
Figure 2.3.7. Discharge Burnup of 25% Power Density Core vs. Pitch.

Table 2.3.2. Scoping Burnup and Isotopic Results.

Case I.D.	Pitch (cm)	Energy (GWd)	Cycle Life	U-235 (Kg/GWd)		U-238 Fiss (Kg/GWd)	Pu (Kg/GWd)	
				Fiss.	Disch.		Fiss.	DISCH.
3A	1.25	1737	1y 316d	0.6343	0.6319	0.0805	0.3291	0.4064
3A	1.60	1849	1y 360d	0.7073	0.4779	0.0495	0.2686	0.2726
4A	1.25	1893	8y 049d	0.6218	0.5293	0.0854	0.3519	0.3895
4A	1.30	2022	8y 252d	0.6210	0.4519	0.0787	0.3512	0.3474
4A	1.35	2033	8y 269d	0.6370	0.4307	0.0708	0.3355	0.3169
4A	1.40	2028	8y 261d	0.6518	0.4195	0.0647	0.3207	0.2969
4A	1.45	2031	8y 266d	0.6587	0.4112	0.0613	0.3127	0.2844
4A	1.60	1881	8y 031d	0.7033	0.4604	0.0499	0.2676	0.2610
8	1.25	1511	8y 043d	0.6387	0.4785	0.0707	0.3227	0.3795
8	1.40	1362	7y 116d	0.7092	0.5386	0.0545	0.2688	0.3448

Note: Ave. fuel enrichment is 2.6% for all cores except for core 8 it is 2%.

Total energy generated assumes a core with 94.42 MTU loading.

Cycle life is calculated assuming 75% availability.

For Core 3A, pitch = 1.2573 which is the Westinghouse value for a standard core using 17X17 assemblies.

Data obtained from LEOPARD burnup calculations.

means that the isotopic and spectral effects occurring during the life of the core cause a shift of the EOL reactivity, favoring less moderated configurations. The main reason for this effect is a conflict between optimum moderation and enhanced conversion ratio: Conversion ratio is enhanced when the resonance absorption in U-238 is increased, but this is obviously an effect which reduces the immediate reactivity of the core. The trade-off appears when realizing that at E.O.L., a large proportion of the power is generated by the converted plutonium. Thus, if conversion ratio is reduced in trying to optimize moderation (and thus increasing BOL reactivity), the core reaches EOL with a reduced amount of plutonium, and therefore becomes subcritical at a lower burnup level.

Note on Table 2.3.2 the relatively high level of burnup achieved by core 8, in spite of being fueled with only 2% enriched fuel. The effect must be attributed primarily to the different moderator characteristics, in addition to the effect of the reduced power level.

It appears at this point that the best pitch from the burnup viewpoint is larger than the one used in the standard reactor, but smaller than the one yielding maximum reactivity at BOL. However, the model used for these scoping calculations can not be assumed to accurately represent a reactor in an equilibrium cycle since BOL compositions with no burned fuel were employed. The question is then whether the optimum burnup pitch obtained in these studies will still be the best for an "equilibrium cycle" core or if it

will be a different one. In order to answer this question it is necessary to perform burnup calculations with a model that can reasonably represent a core in an equilibrium cycle, i.e. there must be some burned fuel present at BOL for each reload core. These scoping studies, however, can serve as guidelines for "equilibrium cycle" calculations.

There are a number of criteria that can be considered important when examining the results of a burnup analysis from an isotopic or fuel utilization point of view:

a). Search for the maximum energy output from the same load of fuel.

b). Search for the minimum discharge of plutonium, for non-proliferation purposes.

c). Search for the minimum amount of net fissile material being consumed per unit of energy produced. This would account for the fissile species being discharged with the spent fuel, and therefore would assume a spent fuel reprocessing policy.

d). In the same way as the criterion above looked for the best resource utilization, it might also be important to look for the best economical combination of resource utilization and cost of fuel cycle (including fuel enrichment, fabrication, storage, reprocessing, etc.).

Although the criteria that involve reprocessing appear more meaningful from the scientific or technological point of view than those which do not include it, the present political situation in the U.S. prevents commercial reprocessing, and this must be

considered when comparing results of burnup or uranium resources utilization.

Since the present study assumes all fresh fuel to have the same characteristics, increasing the discharge burnup implies improving ore utilization. In this aspect, core 4A has a better fuel utilization than core 3A. Although core 8 is able to achieve a high burnup for a low-enriched fuel, the total use of U-235 per unit of energy produced appears to be somewhat worse than that of cores 3A and 4A. However, core 8 is not starting from the same type of fuel, and therefore this result should not be considered as a negative point for core 8. In addition, core 8 could be used to further burn the fuel that is ordinarily discharged from standard PWR's which would in fact represent an improvement of overall ore utilization if fuel reprocessing is still not being considered.

Figure 2.3.8 shows the use of U-235 per unit of energy produced, and the relative amounts of plutonium-to-U-235 burned for the different cores, and for some varying pitches. Note that the amount of plutonium burned is maximum for the tightest pitches, due to the better conversion ratio associated with the hard-flux, undermoderated configurations. As a logical result, U-235 consumption per unit energy produced tends to increase with pitch. It is interesting to realize that for a slight increase of pitch above the standard (from 1.25 to 1.3 cm), the U-235 specific consumption actually decreases slightly. This is due to the reactivity enhancement which occurs in going to larger pitches. The poorer plutonium production quickly offsets this factor and the
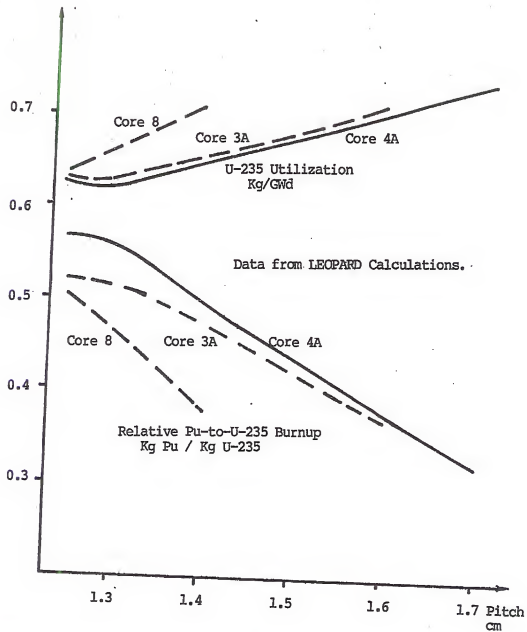
Figure 2.3.8. U-235 and Plutonium usage.vs. Pitch.

U-235 specific consumption begins to increase. It is thus seen
that both U-235 specific usage and plutonium relative burning
remain fairly close to the optimum values for the pitches that
allow highest burnup and therefore longest cycle times, while
worsening rapidly after the region of interest.

2.4. Scoping Work Conclusions

The scoping calculations show that there is good hope for a
significant increase of discharge burnup from the same initial fuel
by using low power density cores. The main effects contributing to
the necessary extra reactivity are the reduced level of xenon and
the reduced effect of Doppler broadening of resonances. A 9%
increase of burnup seems to be achievable by going to a 25% power
density core, while up to 16% improvement appears feasible by
varying the cell pitch. However, these calculations were performed
with a calculational tool which is not well-suited for the burnup
analysis of a core consisting of fuel of various enrichments and
exposure levels. Some changes are surely to be expected when using
an adequate core model for these burnup calculations. The burnup
calculations made in this scoping study are indicative of a
first-load core behavior rather than of an equilibrium core; the
latter is the configuration of most interest for burnup or economic
studies.

It has become evident that a maximum level of automatization
is necessary for the calculational techniques to be used in
detailed burnup calculations. The adoption of LEOPARD to
substitute the chain of PHROG – BRT – MONA – BURNUP proved to be

most advantageous, and a similarly automated scheme should be used for the detailed calculations. However, a new burnup calculational scheme must be developed which allows the adequate representation of a several-batch reactor core in its equilibrium cycle.

After these scoping studies are completed, the in-depth study should be centered on the cores that can best conduce to short-term, commercially feasible low power density systems. The in-depth study is, therefore, centered on a single-core concept having a reduced power density in the 25 to 60% of the standard core. The cores selected for the in-depth study will be discussed in Chapter IV.

The safety aspects of the low power cores are extremely favorable. The fuel average temperature is drastically reduced, indicating a reduced risk of pin damage. The heat transfer conditions have their safety margins, e.g. the DNBR, significantly improved, which makes the probability of critical heat transfer conditions much lower than it is in the standard reactor.

3.1. Burnup Calculational Methods and Present Needs

Once the philosophy behind the development of the SHARP is established and a power density or a range of power densities are chosen, the key work to be done in the area of fuel utilization is the optimization of the pin-cell of the SHARP. The fuel utilization and performance (burnup and isotopic results) are then to be compared with those obtained from a standard power density core to determine its relative merit. This optimization and comparison require a large number of reactor life calculations, where items such as batch-wise burnup, discharge burnup and physical cycle life, etc. are of interest among others.

Many different organizations are interested in developing and performing reactor fuel cycle and burnup calculations. The wide range of organizations and specific interests has caused the development of different calculational schemes, in order to accomodate the different cost/effort/accuracy ratios desired for each particular type of study. Some of the more important types of burnup studies are the following:

a). Vendor burnup studies and core optimization, performed as design work. These studies are most complete, and they generally include three-dimensional, pin-by-pin studies which are closely

coupled with thermo-hydraulics models. These type of studies require large computational efforts and highly specialized personnel, which makes them extremely costly.

b). Burnup studies performed by utilities, either for core following or for licensing. These also have to be detailed studies, because they require strong economical decisions affecting the core cycle, or have significant safety considerations affecting the licensing procedure. However, there is presently a trend by utilities to use more empirical codes (such as nodal codes) which run at lower costs than the fully detailed pin-by-pin studies. Nodal codes generally need base detailed diffusion theory calculations for normalization purposes, but still result in an overall lower cost than the fine mesh neutron diffusion theory codes, while being able to maintain the information necessary for adequate core management.

c). Studies performed by engineering companies or government agencies for assessment of energy policies or new reactor concepts or fuel cycles. These studies fall in a whole new class. They are not directly concerned with the behavior of each particular pin, not even of assembly-wise details. They look at overall results of burnup, time scales, and monetary volumes. Although they need to yield reasonable results, the numerical results themselves are normally less critical, and emphasis is placed on the comparative results of two or more different concepts. These studies generally use some type of empirical formula or simplified core model which

does not require detailed specification of compositions and geometries, as the two previous types of studies did.

d). Burnup studies performed in universities or research-oriented organizations for developmental studies of new reactor concepts, for preliminary reactor design or fuel management. These studies may range in their specificity anywhere from the very rough empirical formulas used for fuel resources utilization and policy studies to a level close to the vendor or utility calculations. However, they tend to be more concerned with particular theoretical details, which call for calculational methods specific to each study. These studies normally follow standard industrial calculational procedures until a certain point where the particular aspect of the study calls for a specific calculational tool, which is often developed for the occasion.

A wide range of other types of burnup or fuel cycle studies exist, but the four categories stated above illustrate the different levels of accuracy or detail that may be sought depending on the purpose of the study, and how they require different calculational costs, manpower needs and calculational tools.

For every burnup study (except for the very simplest ones, where just an empirical formula is applied), two main calculational steps are performed. Each of these steps can have different levels of sophistication depending on the particular needs of the study, and in some cases one or both of the steps may need to be repeated iteratively in order to obtain the accuracy required by the study. Figure 3.1.1 shows the basic flow of data between the two steps.
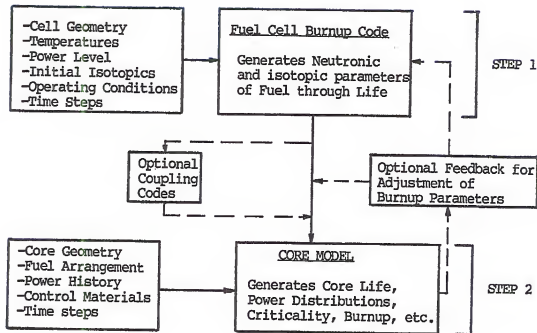
Figure 3.1.1. Basic Steps in a Burnup Calculation.

The first step involves using a code in which a representative unit cell of the reactor is burned. Macroscopic core geometry is avoided; the emphasis is placed in burning a unit fuel cell and surrounding it with neutronic conditions as close as possible to those to be encountered by the "average" fuel element during the real core life. The purpose of this calculational step is to obtain neutronic characteristics of the fuel as it undergoes burnup, which will then be used in the second step of the burnup calculations.

Input to these first-step codes normally includes the following:

a). Fuel cell geometry and initial isotopics.

b). Temperatures of fuel and moderator.

c). Power density (normally linear power density).

d). Definition of a buffer region surrounding the fuel cell in order to modify the neutron spectrum for the presence of structural materials (guide thimbles, fuel assembly cans, spacer grids, water holes, etc).

e). Specification of a series of burnup timesteps that will determine the points where the code has to recalculate the neutronic status of the fuel cell, which has been modified by the burnup process.

f). Specification of controllable poisons present in the fuel cell. This is normally specified as soluble boron concentration in the moderator, and may be specified for each of the burnup timesteps defined for the calculation.

g). Some models may allow for the specification of variable power levels for the different timesteps, the specification of Dancoff factors or some geometric data that may allow the code to calculate them, the specification of a perpendicular buckling, or a buckling search in order to obtain a critical spectrum, etc.

Most of these first-step pin cell codes are self-chaining, in the sense that the user specifies the cell geometry, isotopics and burnup history at the beginning of the code execution, and the code performs the burnup steps and the cross section evaluations after each timestep without user intervention. Other schemes, like the one used for some of the preliminary calculations in the SHARP study, involve several codes to perform the pin cell burnup, and they need the user to manually handle the cross section libraries and the burnup isotopics back and forth between one code and another. The advantages and disadvantages of each method rely mainly on the flexibility and freedom of data handling and geometry specification versus convenience and man time requirements. Figure 3.1.2 shows some possible schematics of the data specification for this first step of burnup calculations.

The output of these first-step codes normally includes the following data:

a). Microscopic cross sections for each isotope present in the fuel cell, for the number of neutron energy groups specified in the input (normally between two and five groups) and for each specified timestep.
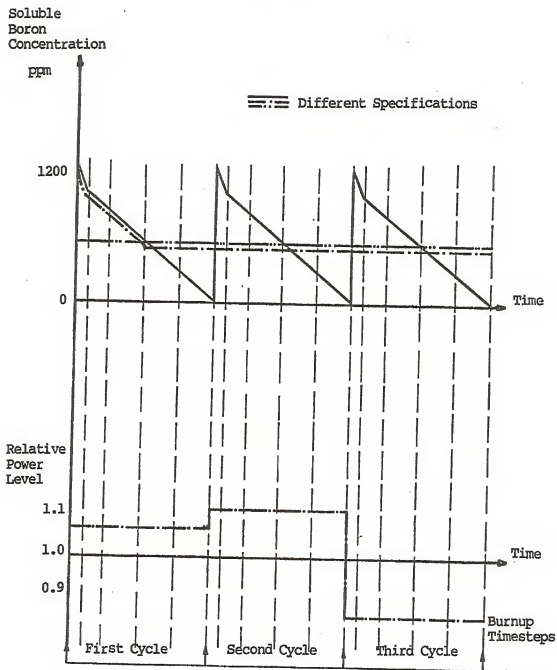
Figure 3.1.2. Examples of Burnup Conditions and Timestep Specification for Fuel Cell Burnup Calculation.

b). Macroscopic cross sections of the fuel and the whole cell, weighted with the neutron spectrum present at each timestep, and with the number of energy groups requested.

c). Infinite multiplication factor of the fuel cell, and effective multiplication factor if a perpendicular buckling was specified.

d). Average neutron speed.

e). Concentration of each isotope present in the cell, regionwise and cell homogenized.

f). Cumulative burnup level since the beginning of the burnup calculation.

g). Other data depending on each particular model, such as the calculated Dancoff factors, fraction of the total core power produced by each fissile isotope, cumulative fission densities, conversion factors, etc.

The second step of the burnup calculations involves taking the data generated by the first code and using it in a core model which accepts overall core information regarding geometry and core operation in order to simulate the actual life of the reactor. This step is the one showing the widest variations from one type of burnup study to another. The most sophisticated models are able to follow the core life without further iterations, while others need to feed their data back to step one of the burnup calculation, for a new iteration. The flow of data from step one to two and back depends on the sophistication of each of the models and the burnup data sought from the overall study.

The second step uses such a wide range of calculational approaches or methods, that it is difficult to specify the general input requirements and output data. As a general rule, the explicit core model (in which the fuel is represented either pin by pin or in relatively small nodes, and control materials are specifically treated) needs the initial isotopics information, the microscopic cross section libraries generated by the first step of the burnup calculation, the core geometry data, the power history, the control materials history, and general editing information, while the output includes core isotopics in zone averages, pin by pin, or whatever region type is used by the code versus core history; criticality evaluations, power distributions, burnup information in average or explicit for each unit considered in the core, region averaged multigroup neutronics data, etc. Codes of this type are for example PDQ-7(17), SIMULATE(18), etc.

Figure 3.1.3 shows the geometry and material zones specifications needed for a two-dimensional calculation of a quarter assembly of a PWR with PDQ-7. Note the extreme detail of the geometrical description, and the relatively large number of different compositions considered. It is easy to imagine the extreme complexity of the calculation of a whole core in three dimensions, with the necessity to identify different fuel batches, control rod assemblies, burnable poison rods assemblies, assemblies with water holes, etc. Nodal codes avoid some of the complexity and reduce computational time by lumping each fuel assembly into a few nodes, but they require node-interaction parameters and albedos

_____ Diffusion Calculation Mesh

▭ Fuel Pins

▨ Interassembly Water Gap

▨ Water Holes (or LBP's)

▨ Instrument Thimble
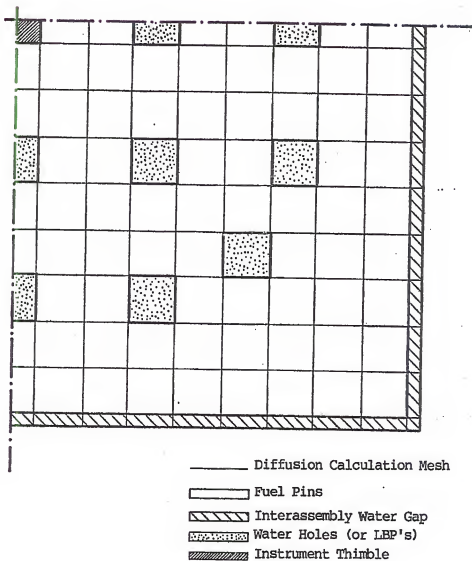
Figure 3.1.3. Geometry and Composition Specification for a PDQ-7 1/4-assembly Burnup Calculation.

which must be normalized with a detailed core calculation, as well as requiring adjustment of a large number of empirical factors.

There are other computational models which run at lower costs and may not require the technical expertise needed for running full blown PDQ-7 models or three-dimensional nodal codes. These models normally accept a lower degree of geometrical sophistication, or they may do just a part of the burnup calculation. For example, they may perform the criticality and eigenvalue calculations, but not be able to perform the actual core burning, which then has to be done by a separate code. This is the case of MONA, which was used in some of the preliminary studies of the SHARP. This type of codes are useful for criticality studies or spectral effects analysis, but they are of little use for a real life burnup study because of the enormous manual data handling required. However, they have the advantage of the low cost and the relatively simple input.

Other types of calculational models are often used for scoping or medium accuracy burnup calculations. They normally are simplified codes which use part of the data obtained from the first step of the burnup calculations, and then introduce an approximate core model, using empirical factors or formulas whenever the available data is insufficient for a rigorous treatment of the items needed for the calculation. However, these types of codes cover a wide range of sophistication and thus their accuracy and the information that can be obtained from them is largely variable. The core model used for the SHARP can be included in this

classification, evolving from a well known calculational model. The innovative core model used for the SHARP is an improvement on the accuracy of the results, resembling closer the real reactor life. It is achieved by the proper handling of significant core history data, which appear to have been previously neglected. An explanation of the data handling by these codes follows.

These simple calculational models are normally based on a pin cell burnup calculation and often use the infinite multiplication factor calculations in order to obtain an estimate of the core criticality state or its reactivity.

In the simplest model, a pin cell would be burned for its whole life without any poisoning, and at a constant power level. The characteristics of a pin are a good representation of the fuel batch to which it belongs, since the pin is surrounded by many fuel pins alike. The only exceptions are the pins facing a fuel assembly of a different batch (and therefore having a sharply different burnup status) or facing the reflector. Then, if the fuel pin is burned for its entire life, it certainly passes through the stages corresponding to each batch in the core. This means that it should be possible to take some of the neutronic properties of the pin being burned in the cell burnup code and obtain the characteristics of each batch present in the core at any particular time, which should allow in some way to calculate the status of the whole core. If the core contains three batches, the cell burnup calculation should be performed in such a way that the timesteps chosen would be repeated three times, and each of the three sets of

timesteps would span exactly the expected duration of one core cycle. This would allow the data from the corresponding timesteps in each cycle to be grouped into a core timestep. Figure 3.1.4 illustrates this "timestep synthesis" from a hypothetical cell burnup scheme in which five timesteps were allowed for each of the three cycles of core residence of the fuel. The data sought is the infinite multiplication factor of the core, based on the unpoisoned multiplication factor of the pin through its life. The figure shows the conceptual grouping of each timestep's data, but not the calculations involved in it, which will be dicussed later. Note that the timestep pattern is repeated in each cycle of the cell burnup scheme, in order to make the timesteps correspond to the same time of core life.

However, there are several ways of specifying the pin cell burnup and of doing the calculations for collapsing the cell timesteps into core life timesteps. Some of the systems used for pin cell burnup specification will now be examined.

The simplest scheme, as mentioned before, burns the fuel cell without any poisoning and at the nominal power, for the length of time that is estimated that the fuel will remain in the core. Then the multiplication factors of the cell timesteps are collapsed in order to obtain the core multiplication factor through the core life. An interpolation or extrapolation of the curve of core multiplication factor as it reaches a value of unity determines the end of the core cycle. The core cycle length is then optionally used to perform new iterations of the fuel cell burnup calculation,
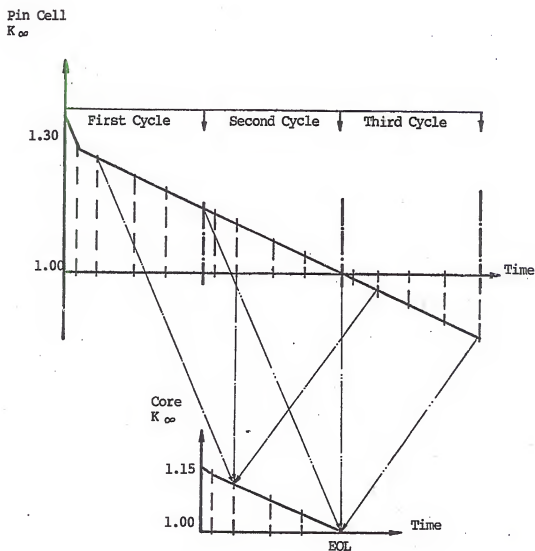
Figure 3.1.4. Grouping of Pin Cell Data into Core Data.

until it agrees with the cycle length estimated by the timestep collapsing procedure.

This scheme, however, involves gross errors because the absence of soluble boron causes variations of some percent in the cross sections of some isotopes such as U-235 and Pu-239. This entails errors in isotope concentrations and as a result, errors in the multiplication factor calculations.

Another cell burnup method, which partially solves this problem, specifies a soluble boron concentration in the moderator with a value similar to the time-averaged boron concentration of a real core. This concentration is approximately 400 ppm. But then the fuel cell multiplication factors can not be directly used for the calculation of the core multiplication factor, because they involve the soluble boron poisoning. The problem is solved by using a very short "pseudo-burnup" timestep, with no soluble boron, after each real burnup timestep. A "pseudo-burnup" timestep is a very short timestep which practically adds no burnup to the fuel, so that the isotopics are not altered, but still allows for the eigenvalue calculation in the boron-free configuration. This way, the fuel is always burned with a spectrum influenced by the 400 ppm of soluble boron, but the multiplication factors are obtained from the pseudo-burnup timesteps which contain no boron, and the "clean" batch reactivity can still be obtained through the fuel life. Figure 3.1.5 illustrates this method, showing the boron specification and the multiplication factors that may be obtained through the core life.
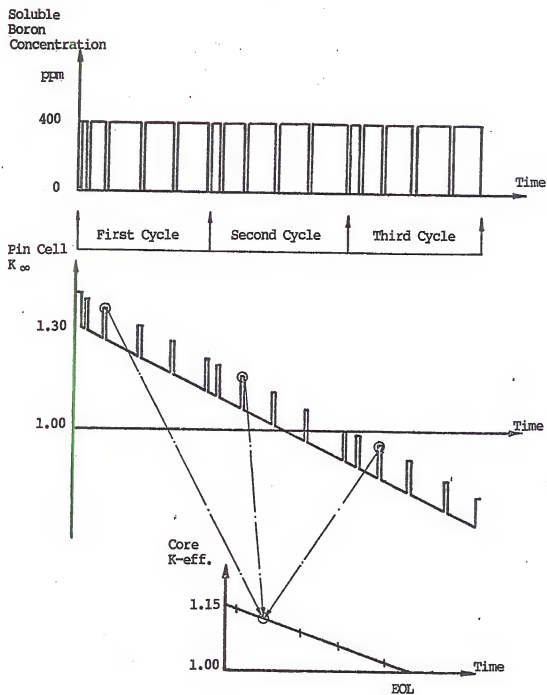
Figure 3.1.5. Constant-boron Pin Cell Burning with Boron-free Pseudo-burnup Steps.

The advantage of this method is that the core isotopics do not contain the heavy errors due to errors in the neutron energy spectrum, and that the method is still rather simple. On the other hand, the pin cell burnup calculations require a much larger number of timesteps, which increases the computational costs, and although the final isotopics may not be grossly wrong, there is no guarantee that the chosen boron average concentration is really the average for the real reactor life.

PDQ-7 assembly burnup calculations conducted have shown that an average error of 50 ppm in boron concentration (even in real-time concentration following calculations) causes isotopic errors of 1.2% in Pu-239 and 1.27% in Pu-241 for a burnup level of 34000 MWd in a fuel cell configuration of a tight 1.20 cm pitch (containing very small amounts of moderator and its associated soluble boron). For a large pitch cell of 1.60 cm, isotopic differences of 3% in Pu-239 were found after only 9000 MWd burnup, for an average error in boron concentration of 100 ppm. These facts require a very precise specification of soluble boron concentration if accurate isotopics are needed, and this is the case if one is concerned about core life determination. The system described on Figure 3.1.5 has the further disadvantage that the fuel is burned without the critical spectrum, and although the discharge isotopics may not be grossly in error, the errors existing at any particular time during the core life are unknown, and so are the errors incurred in calculating the core multiplication factors, which in turn will determine the core life.

The above reasons induced the author to consider that the specification of the boron level that would keep the reactor critical at all times is of significant importance for an accurate determination of the core life and should not be neglected, as is commonly done in simplified burnup calculations.

Following the soluble boron concentration through the life of the fuel implies a fuel cell burnup boron specificaton such as the one shown on Figure 3.1.6. This solves the problem of the erroneous neutron spectrum, but creates the new problem of the boron letdown curve needing careful adjustment, mainly concerning the core life duration. This is needed because specifying too short or too long of a cycle time on the boron letdown curve could cause the period of fuel burning to be done with a very low boron concentration when the highest concentration would be required, or vice versa, and this would bring back the problems of erroneous isotopics and eigenvalue estimates because of inexact spectra. This boron determination requirement is solved by the calculational method described later in this chapter. This new method contains important differences with the schemes described above. The previous schemes relied on an extrapolation or interpolation of the core multiplication factor (or sometimes on the core reactivity, which is supposed to follow a more linear behavior) for the determination of the core life. The scheme used in this work, where soluble boron is constantly adjusted to its real value, can not rely on these eigenvalues since the core eigenvalue should by definition be kept at a value of unity throughout the core life.
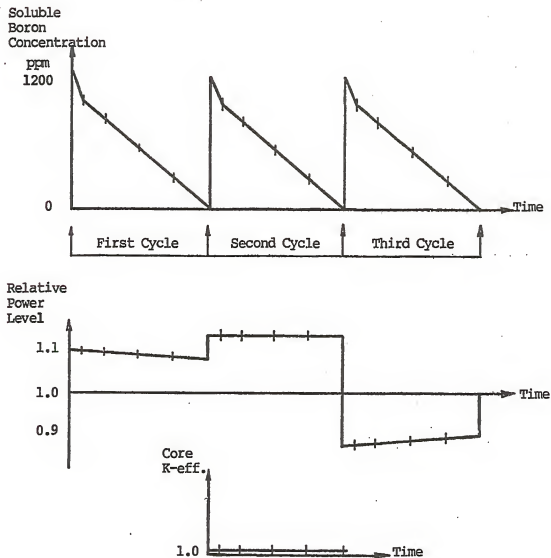
Figure 3.1.6. Pin Cell Burning with Soluble Boron Letdown Curve and Power Level Following.

It would still be possible to use the "clean" pseudo-burnup steps, but the author decided that the adjustment of the boron letdown curve to the point where its value should be zero was best suited for this purpose. The advantage of using the boron curve for the EOL estimate is that it contains the constantly varying spectral and self-shielding changes, which depend on core burnup status and boron concentration, while other popular models neglect these aspects.

The above paragraphs discussed the need for an accurate determination of the soluble boron concentration level in the fuel cell burnup code, but almost as important as the boron is the specification of the true power density level through the fuel life.

It is frequent to see that the simple burnup schemes do not specify changing power density levels for the fuel as it progresses through burnup. It is even frequent to see almost an equivalence being used when referring to real-time core life and burnup since they are considered to be so closely equivalent. However, it is well known that the different batches in the core do not share equal power levels in spite of the efforts to flatten the power distributions. As a general rule, the batches being burned for the first and second cycles generally hold a higher power level than the third-cycle batch.

When assigning a boron letdown curve for the fuel cell burnup, one is not only assigning a certain boron concentration to a core life-time, but also to each burnup point through the fuel life.

This means that a certain boron poisoning is assigned for each isotopic status of the fuel. Figure 3.1.7 illustrates the consequences of not specifying the different power levels of the fuel as it advances through its life.

If the power level is maintained constant, as is the case with most of the simple burnup calculational schemes, the end of core cycles correspond to 1/3 of the discharge burnup, 2/3 of the discharge burnup, and the total discharge burnup, in a three-batch core. Thus, the burnup levels slightly after 1/3 and 2/3 of the discharge burnup are burned with a very high boron level, which corresponds to the beginning of the core life. However, the real core situation has a higher burnup at the beginning of the second and third cycles, because of the higher power level of the fuel during the first two cycles of core residence. This burnup-time mismatch of the flat-powered model as compared to the real core burnup situation may cause a sustained error in the correspondence of soluble boron to burnup level, as shown on Figure 3.1.7, with the resulting errors in batch multiplication factors and neutron spectrum.

Figure 3.1.7 shows a hypothetical boron letdown curve for a core cycle, and using the same time axis, the fuel burnup is represented on the ordinates for both a case where the power level is kept constant and for a case where cycles 1 and 2 have a somewhat larger power level than cycle 3, as is the case in a real core. It is possible to see the sustained boron concentration mismatch, which becomes most accentuated in the vicinity of the
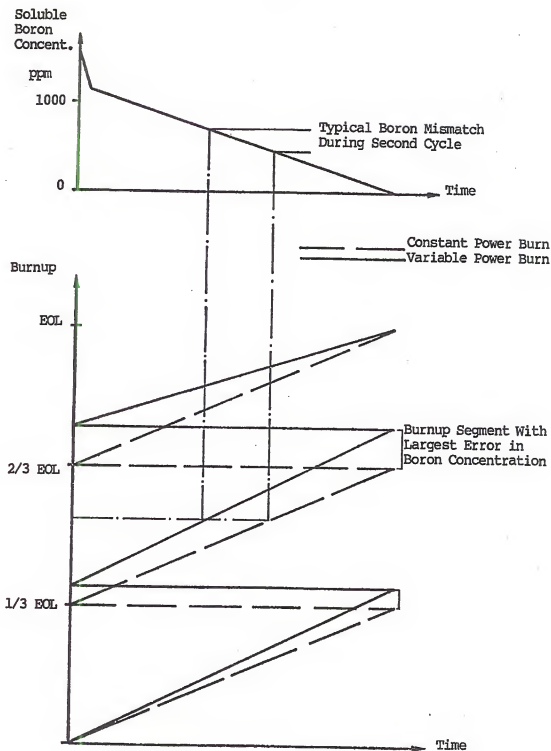
Figure 3.1.7. Effect of Time-dependent Power Level and Soluble Boron Concentration Specification.

cycle ends, due to the fact of having the two burnup schemes start their second and third cycles with different burnups.

This boron mismatch causes yet a further problem: at the BOL, the constant-power scheme forms the core with batches having burnups of 0, 1/3 discharge and 2/3 discharge. Meanwhile, the varying-power scheme starts the cycle with burnup of 0 for the first batch, somewhat more than 1/3 of discharge burnup for the second batch, and somewhat more than 2/3 of the discharge burnup for the third batch. These increments are of the order of 5% of the burnup level for the first batch, and of the order of 8% of the burnup level for the second batch, which represent non-negligible differences in the batches' reactivities. Having a higher level of burnup in two out of the three batches, the varying-power scheme presents a lower reactivity than the constant-power scheme, thus requiring a lower level of soluble boron in order to achieve criticality. This situation is maintained, with different sizes of mismatch, through the whole core cycle, thus causing evidently a different critical boron letdown curve for each of the schemes, and yielding a necessarily different core cycle length. This, together with the spectral effects of soluble boron are the two main causes that require maintaining the real power levels during the fuel cell burnup calculations.

Once it is clear how the simplified burnup models can use a fuel cell burnup scheme with its chained timesteps and all the different adjustable data in order to obtain the core behavior through life, it is necessary to look at the different methods

available for combining the several fuel cell timesteps that form the complete core at each particular time.

The core is composed, at any timestep of its life, of a certain number of fuel batches. Each of these fuel batches is represented in the simplified burnup core model by one timestep from each of the cycles specified in the fuel cell burnup calculation, as shown on Figure 3.1.4. The problem is, at this point, to determine the core multiplication factor from the multiplication factors of each batch. This can also be done in a number of ways, with varying degrees of complexity. The simplest model gathers the batches' infinity multiplication factors and averages them. The result is considered to be the multiplication factor of the core. That is,

$$Kc = \frac{(\Sigma_i \ Ki)}{n}$$

where Kc = Core effective multiplication factor.

Ki = Infinite multiplication factor of batch "i".

n = number of batches in the core.

Note that the fuel cell burnup schemes often include a perpendicular leakage factor, so that what is referred here as "batch infinite multiplication factor" may well be already corrected for axial leakage, which makes the core multiplication factor appear much more realistic when compared to the more sophisticated core criticality calculations.

The above estimate of the core multiplication factor, although frequently used for rough criticality estimates, is rather simple-minded, since it does not take into account the fact that there are batches which for their combination of power sharing and geometrical situation represent a heavier weight in the core multiplication factor. It does not take into account either the fact that the geometrical disposition of the batches in the core causes some batches to experience a much larger radial neutron leakage than others.

More complex type of "simple" burnup calculation core models use weighting factors from the batches' power sharing in the calculation of the core multiplication factor. For example,

$$1/K = \Sigma_i (Fi/Ki)$$

where K = Core effective multiplication factor.

Fi = Fraction of the core power produced by batch "i".

Ki = Multiplication factor of batch "i".

In this case, the power fraction generated by each batch in the core may be obtained as

$$Fi = \frac{Ki^n}{(\Sigma_i Ki^n)}$$

This calculational scheme is significantly more complete than the simple K-averaging one described before, but still does not take into account the very important fact that one of the batches

lies on the core periphery, thus experiencing a radial leakage which is quite different than that of the internal batches. There is a model which corrects for this effect by multiplying the peripheral batch multiplication factor by an empirical non-leakage constant, which is again an improvement of the core model. However, the radial leakage of the peripheral batch is mainly affected by the reflector characteristics, and since the reflector's main component is cooling water, the neutronic changes experienced by the coolant during the core life must heavily affect the reflector's performance. Indeed, the changes in boron concentration in the water from B.O.L. through E.O.L. cause the thermal absorption cross section to vary by a factor of about three, and even the transport cross section is affected by some percent. This suggests that if soluble boron concentration is important for the spectral effects in the core, it is also important in its effect on the reflector characteristics which directly affect the radial leakage, and this effect should be also taken into account if at all possible. It appears that this effect has not been taken into account in known simplified burnup models; it has been included in the model used in this work.

Whenever a simplified model for the representation of a complex system is used, it is necessary to reach some compromise and take into account only those factors which are considered of first importance, while bypassing others of lower importance. The factors considered of importance in the model used in this work are summarized in the following specifications:

a). The fuel burned in the "basic" codes must be imbedded in a flux spectrum which is as close as possible to the one existing in the actual reactor for the fuel being studied.

b). Boron concentration must be kept as close as possible to its true value throughout the core life, in order to avoid erroneous spectra that would affect both the criticality calculations and the isotopic evolution.

c). The model must be able to follow a "real time" evolution of the core, as opposed to using burnup as a time measurement, because burnup is not uniform through the core life, and some isotope appearing or being burned earlier or later in the core life affects the spectrum, the criticality study, the isotopics, and eventually the cycle length.

d). In order to accomplish the previous point, the model must be able to assign the right power to the fuel at each point in life, i.e., if a "real time" model is sought, it is essential to have a correct time-power-soluble boron correspondence.

e). In order to obtain the correct batch power assignment at all times during the fuel life, it is necessary to have a fair estimate of neutron non-leakage probability for each batch, which if at all possible, should be made automatically adjustable through life.

With all these ideas in mind, the model was developed as described in the next sections, starting with a fuel pin code, going to a fuel assembly burnup calculation, and finally feeding the data to the core-simulation code.

3.2. Method Developed for this Study

3.2.1. The Fuel-burning Codes.

As it was explained in the preceding section, any burnup calculation must start with a fuel-burning scheme which provides the necessary data for the subsequent core model. In the case of the SHARP study, the number of initial calculations needed was rather large, due to the significant number of different power levels and pitches that had to be studied. This required a rather automatized calculational procedure which should be of a relatively low cost. In addition, since the differences between the several cores to be studied might not be too large, calculations need to retain maximum accuracy while again not incurring in excessive costs. The first task was to choose the code or codes to be used for the fuel-burning step.

One of the more classic pin-burn codes is LEOPARD(11), which is described in more detail in Appendix B.5. LEOPARD uses a MUFT-SOFOCATE(12,13) scheme for the calculation of cross sections for a pin-cell geometry, and it has been and is being widely used in industry for many of the burnup study types described on the preceding section. LEOPARD offers flexibility, low cost and simple input requirements; however, LEOPARD presents some problems for the SHARP study:

a). It does not allow for a change of the pin power during the burnup process, thus making specification 4 of the model requirements hard to keep.

b). It has known restrictions concerning the self-shielding calculations. These restrictions might be detrimental in the SHARP study since the somewhat high burnups expected entail higher-than-normal concentration of highly absorbing nuclides such as fission products and plutonium.

The EPRI-CELL(19) code was also considered. Although EPRI-CELL is not a "classical" code yet because it is fairly new (released to industry users around 1977), the calculational scheme it uses is based on well-proven calculational methods, and the benchmarking done to date proves its high accuracy. EPRI-CELL uses a GAM-THERMOS-CINDER(20-22) scheme for cross section evaluation and fuel burning isotopic chains. It presents basically all the advantages of LEOPARD (namely, it is a pin-cell code of simple input and automatically chains spectrum and burnup calculations) and it does not present the two limitations of LEOPARD, thus allowing a higher flexibility and accuracy in the pin-burn calculation. The only disadvantage of EPRI-CELL when compared to LEOPARD is its running cost, which is about one order of magnitude higher. Despite this drawback, EPRI-CELL was chosen as the pin-burn code, with the initial intention of using it as the only calculational tool external to the core model.

However, when examining the documentation of the EPRI ARMP code package (of which EPRI-CELL is a part), it is evident that the purpose of EPRI-CELL is to generate sets of cross-sections through the life of the fuel for use in generating the HARMONY tables for PDQ-7. PDQ-7 is then expected to be used for the actual core-life

calculations, since it allows for multi-dimensional and complex geometry specifications. PDQ-7 calculations are long and cumbersome if one is trying to model a two or three-dimensional large grid, but it may be used with reasonable ease for a small geometry (a 1/4 fuel assembly in two dimensions, for example).

This last fact suggested that PDQ-7 could be used instead of EPRI-CELL in order to perform the fuel burnup calculations. This may appear awkward since it first takes an EPRI-CELL run to feed the cross section tables to PDQ-7. The system requires more codes in order to obtain apparently the same data for the core model, and manual data handling with PDQ-7 is far more cumbersome than with EPRI-CELL, even for a simple geometry case. However, there are a number of advantages in using PDQ-7 instead of EPRI-CELL for the fuel-burning calculations. Some of these advantages are given below:

a). Given the type of core model that must be used for the burnup calculations of the SHARP study, it normally takes more than one iteration to adjust all the core life parameters described in the previous section (soluble boron letdown curve, batch relative powers, etc.) and therefore, each case studied requires several fuel burnup calculations. If PDQ-7 is used, the iteration only requires repeating the PDQ-7 calculation and not the EPRI-CELL one, if the fuel cell characteristics are not altered. Each 1/4 assembly PDQ-7 calculation costs only one fourth as much as an EPRI-CELL one, because PDQ-7 is a diffusion-theory code working with pre-tabulated few-group cross section libraries, as opposed to

EPRI-CELL, which is a neutron transport theory code, working with raw fine-group libraries. However, PDQ-7's calculations may be as accurate or even more accurate than EPRI-CELL's as will be explained below.

b). For a well-versed user, PDQ-7 has almost no restrictions. The code allows for extraordinary flexibility in the specification of the parameters for simulation of the core environment for the fuel burning process, and it has very powerful editing capabilities.

c). EPRI-CELL simulates the reactor environment affecting the fuel pin by placing a buffer region around the moderator ring. This buffer region is composed of coolant and a certain proportion of structural materials which modify the fuel cell neutron spectrum. This system yields reasonably accurate average isotopics for the fuel batches, but there are really very few pins in any assembly which would behave as a "batch average" pin. PDQ-7, on the other hand, can specify the geometry of a 1/4 fuel assembly as described on Figure 3.1.3 in the previous section. That description treats each fuel pin in the assembly as a separate unit, and it is obvious that the pins neighboring water holes or facing the interassembly water gap do not see the same spectrum as a pin surrounded by identical pins. These spectral differences among fuel pins may translate into slight differences in the calculation of batch-wise multiplication factors due to the combination of different spectra and different isotopic composition of each pin in the fuel assembly. If this is the case, PDQ-7's

estimate of the batch multiplication factor should be closer than EPRI-CELL's, because of its more realistic modeling of the really repeating pattern in the fuel batch, which is the fuel assembly, rather than the fuel pin.

It should be pointed out that the use of "simplified" geometries for the PDQ-7 calculation could result in severely distorted results. The 1/4 assembly requires the specification of different compositions due to the presence of the water holes and instrument thimble, and one might be tempted to use a simple 2X2 pin geometry, trying to take advantage of PDQ-7's flexibility while avoiding the additional data handling complexity of the complete 1/4 assembly. The results of a 2X2 pin array burning can be just disastrous. Not only it does not take into account the differential effects of the unevenly distributed water holes and interassembly gaps on the different fuel pins of the assembly, but being composed of just fuel element cells, it does not include any material simulating the effects of the structural materials, which in EPRI-CELL are represented by the buffer region. Such a burnup calculation may result in isotopic errors of the order of 13% for U-235 and 20% for Pu-239 at a fuel burnup level of 40 GWd/MTU, with the associated errors in the calculation of batch multiplication factors.

When PDQ-7 is finally selected for the fuel-burning calculation, a third code from the ARMP package needs also to be used in order to simplify the point-burnup calculation. This code is NUPUNCHER(23) and its function is to translate the

burnup-dependent cross sections generated by EPRI-CELL into a card deck with an adequate format for the HARMONY(24) part of PDQ-7. The deck includes the description of the isotopic chains and their constants. It also allows for the cross sections to be fitted against any variable that can be deemed significant on any given cross section, for any particular isotope (for example, Pu-240 thermal absorption cross section is generally not expressed as a function of the pin burnup, but rather as a function of Pu-240 concentration itself). NUPUNCHER greatly simplifies the task of preparing input to PDQ-7.

Once the quarter-assembly geometry is set in PDQ-7, it is necessary to burn the fuel following the circumstances encountered in an actual reactor core as closely as possible. One of the main parameters that influences the evolution of the fuel is the neutron energy spectrum, as was explained above. The neutron energy spectrum existing in the reactor at any given time depends on many factors, some of which are related to the physical design of the core, and therefore are implicit in the geometric and compositional description of the quarter-assembly, while others depend on factors that may vary during the core life. Two of the main variable parameters influencing the neutron spectrum are the soluble boron concentration and the power level. The soluble boron concentration directly affects the neutron spectrum because it is a strong thermal-neutron absorber. The power level affects the neutron spectrum in an indirect way, because it determines the concentration of xenon present in the fuel, and xenon is also a

very strong thermal neutron absorber. Furthermore, the power level
of the assembly through life affects in a significant way the
concentration of most isotopes through the core life, which is a
further reason calling for an accurate specification of the power
level at all times.

Thus, when the quarter-assembly modeled in PDQ-7 is burned for
its complete core life (a series of as many consecutive cycles as
fuel batches are present in the core), an estimate of the soluble
boron concentration versus time is specified for each of the
cycles, as well as the power level of the batch at each specific
moment. A first guess of possible soluble boron concentration and
power level evolutions through the whole life of an assembly are
shown in Figure 3.2.1.
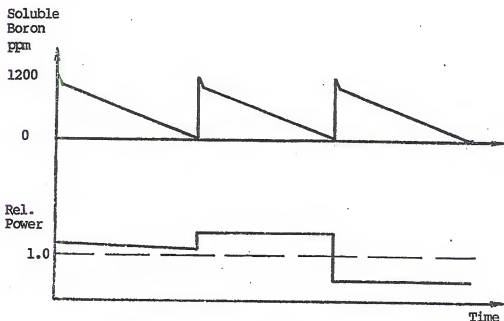
Figure 3.2.1. Estimates of Boron Letdown and Relative Power.

Note that the point in Figure 3.2.1 where the soluble boron concentration reaches a zero value and jumps back up to the value it had at BOL signifies the core cycle duration. Once the assembly is burned according to the power and boron concentrations assumed for all its life, all these burnup environment data together with the multiplication factors of the assembly at each time step are fed into the core model code. The core model code must evaluate the multiplication factor of the whole core, the power level of each batch, and the core cycle life duration; it thus furnishes data for a better estimate of the real behavior of the core through life. If the newly calculated data from the core model code agree closely with the power levels, boron concentrations and cycle lengths previously input to PDQ-7, the guess is final, and burnup, core life and isotopics can be obtained from PDQ-7 and the core model. If the core model calculated data differ significantly from the data input to PDQ-7, it is necessary to modify the assembly burnup data according to the core model, and run a new PDQ-7 burnup case, starting a new iteration.

Figure 3.2.2 shows a flow chart of the codes and data handling necessary in order to obtain the fuel burnup data to be input to the core model code. The core model code is described in the next section.

3.2.2. The CRIBUR Core Model.

The CRIBUR code was developed as a simple model for core life calculations which would include all the points outlined in Section
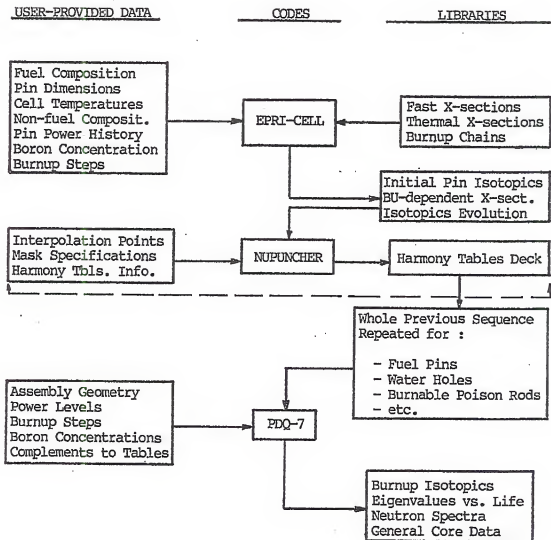
Figure 3.2.2. Flowchart of Data for a Fuel Burnup Calculation Using PDQ-7 1/4-assembly Geometry.

3.1 as being of significant importance for the determination of the core life. Consequently. CRIBUR needs the input of a string of infinite multiplication factors of a fuel batch burned with a scheme in which the soluble boron letdown curve has been specified. as well as the power levels of the batch through its life. CRIBUR calculations include the following:

a). The criticality status of the core at each timestep through its life. by combining the multiplication factors of the batches in the corresponding time.

b). The necessary adjustments to the soluble boron letdown curve in order to maintain core criticality.

c). A new estimate of the core life by interpolation or extrapolation of the soluble boron letdown curve.

d). The power level of each batch during its core residence.

The fact that the core model emphasizes the core criticality through life gave it the name of CRIBUR (CRItical BURning).

CRIBUR considers the core as an infinite cylinder surrounded by a reflector. Radial leakage is obviously not the same for the peripheral batch as for the inner batches, and thus it requires a special treatment; on the other hand, the axial leakage is common to all the batches. and is directly accounted for in the 2-dimensional PDQ-7 assembly calculation. thus making it unnecessary to keep track of it in the core model.

In this study. the core model represents a Westinghouse 3400 MWth. four-loop system with a core radius of 168.53 cm. This

dimension is used for radial leakage calculations, and although it is hardwired into the program, it may be easily changed if a different core radius is considered. Figure 3.2.3 shows a schematic drawing of the geometry used in the core model.
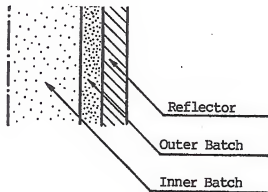
Figure 3.2.3. Core Geometry Used in CRIBUR.

Note that the core is considered as being composed of two fuel regions: an inner zone containing all the fuel batches except the one placed at the core periphery, and a peripheral region containing only the peripheral batch. The reflector then surrounds the fuel region. This configuration is used in order to account for the different radial leakage effects on the different batches. In a large PWR core such as the one being studied, it is reasonable to assume that no significant net radial leakage occurs from the non-peripheral batches, and therefore, all net neutron leakage in the radial direction is associated with the peripheral batch. This neutron leakage of the peripheral batch is accounted for in the core criticality calculations by modifying the infinite

multiplication factor of the batch in the way described below. Since the reflector is composed of cooling water, it contains varying amounts of soluble boron at different points in life, which makes its reflecting characteristics variable with time. This effect is accounted for in CRIBUR as described below.

Due to the fact that the peripheral batch has some neutron leakage into the reflector, its multiplication factor is affected by the probability of a neutron leaking out of the fuel zone. Expressed in mathematical terms, this would be:

$$Kb = Kib * Pnlb$$

where Kb = K of the peripheral batch, after radial leakage modification.

Kib = K inf. of the batch, as obtained from the point-burnup calculation.

$Pnlb$ = Probability of non-leakage of neutrons from the peripheral batch.

By definition of probability, it is possible to write:

$$Pnlb = 1 - Plb$$

where Plb = probability of neutron leakage from the external batch.

And similarly, it is possible to write for the whole reactor:

$$Pnlr = 1 - Plr$$

where Plr = Probability of radial leakage for the whole reactor.

The problem is now to obtain a reasonable value for Pnlb. Considering that the zone occupied by the peripheral batch is very thick in terms of the neutron diffusion length, it is possible to assume that none of the neutrons leaking from the core into the reflector come from the inner region batches, and therefore all net losses are from the peripheral batch population. Since in the present large PWR cores the radial power distribution is maintained fairly flat, it is also reasonable to assume that there is no significant net neutron current between the outer and the inner core zones. In this case, the probability of leakage of a neutron belonging to the peripheral batch can be expressed as the probability of leakage of a neutron from the whole reactor, scaled up by the ratio of population of the whole core to that of the peripheral batch. This can be mathematically expressed as follows:

$$Plr * Nr = Plb * Nb$$

where Nr = reactor neutron population.

Nb = peripheral batch neutron population.

Recalling that the probability of a neutron leaking can be expressed as

$$Plb = Nleak / Ntotal$$

where Nleak = number of neutrons leaked.

Ntotal = neutron population.

It is possible to rewrite the peripheral batch neutron leakage probability as

$$Plb = Nleak / Ntotal = Plr * Nr / Nb$$

This would require knowledge of the total core and peripheral batch's neutron populations. but since the parameter of interest is really the ratio of these populations rather than the individual values of any of them. it is possible to make a further approximation that will simplify the calculation. The approximation assumes that the neutron population in the peripheral batch has approximately the same average speed as the whole core population. This is not completely exact. because the different isotopic compositions of the various batches. and the fact that fast neutrons tend to leak in a larger proportion than thermal neutrons, cause slight differences in the batch averaged neutron speeds but the error incurred is of negligible importance. Once this assumption is accepted, since the neutron flux is

$$\bar{\phi} = n * v$$

where n = neutron density.

v = average neutron speed.

it is possible to express the ratio of neutron populations of the core and the peripheral batch as

$$\frac{Nr}{Nb} = \frac{\sum_i \bar{\phi}_i * Vi}{\bar{\phi}_b \quad Vb}$$

where Vi = volume of batch 'i' in the core.

Vb = volume of the peripheral batch.

Since normally all the batches in the core have the same number of assemblies, all Vi's are equal, and can be factored out of the summation and cancelled with the Vb of the denominator. The probability of non-leakage from the peripheral batch becomes

$$Pnlb = 1 - ((1-Pnlr) * \Sigma_i \, \phi_i \, / \, \phi_b)$$

It is now necessary to find some way of obtaining the probability of neutron non-leakage from the reactor and the batches' average fluxes with the minimum data handling possible, since all these data have to be input to the core model.

There are well known expressions that determine, as a good approximation, the probability of neutron non-leakage from an homogeneous reactor, the reflector effect being accounted for with the use of the "reflector savings". The batches' fluxes can be obtained from the fuel burnup calculations. Considering that neutrons can leak during slowing down or after becoming thermal, the probability of non-leakage can be expressed as

$$Pnlr = (1 + L^2 * B^2) \quad * \exp(-\tau * B^2)$$

where L = diffusion length of the core.

B = geometrical buckling.

τ = neutron age.

However, it is important to note that the above expression used for determining Pnlr is correct when applied to a bare homogeneous reactor. The reflector effect is mocked up by adding

the reflector savings to the core dimensions to obtain an "equivalent" bare system. In any case, the leakage calculation is still based on a homogeneous, cylindrical core whose flux magnitude in the radial direction is assumed to follow a Jo Bessel function. It is obvious that real reactors do not have such a flux shape, mainly because of the geometric disposition of the different fuel batches in the core.

Given a certain total neutron population of a reactor, the probability of leakage must obviously depend on the geometric distribution of the neutrons inside the core. In other words, for a given reactor type, if the neutron population is concentrated near the centerline, there should be a lower escape probability than if most of the neutrons live near the periphery. This consideration suggested the need of modifying the non-leakage probability with a factor that would somehow take into account the geometric difference between the real flux distribution of the core being studied and the theoretical Jo Bessel shape. For this reason the author called this factor the "Shape Factor".

One assumption used in calculating the shape factor is the previous assumption that all radial leakage comes from the peripheral fuel batch. A second simplifying assumption states that the actual radial leakage from the reactor is proportional to the ratio of the neutron population of the outer batch over the population of the whole core. In other words, it is assumed that a reactor having 50% of the total core population in the outer batch presents more radial leakage probability than a core having only

25% of the total population present in the outer batch. The proportionality constant can be easily obtained with a few simple calculations using a diffusion theory code. This procedure gives a simple way of scaling the leakage according to the population or flux distribution. Another aproximation needed (which was also used before) is that of considering the average speed of neutrons being constant across the reactor; this allows the use of volumetric flux averages instead of neutron populations for the leakage calculations.

With these assumptions in mind, the shape factor can be expressed as

$$SF = \frac{(\bar{\phi}_b / \sum_i \bar{\phi}_i)}{(S1 / (S1+S2))}$$

where $\bar{\phi}_b$= actual flux average of the peripheral batch.

$\bar{\phi}_i$= actual flux average of batch "i" in the core.

S1= neutron population of the peripheral batch in the Jo-shaped core.

S2= neutron population of the internal core zone in the Jo-shaped core.

For any given number of batches, S1 and S2 would be constant values, but since CRIBUR was designed to handle cores with any number of batches up to five, it is necessary to express S1 and S2 as a function of the number of batches present in the core, since the peripheral batch will correspond to a varying thickness of the peripheral shell.

If n is the number of batches present in the core, recalling that the first zero of the Jo function is at 2.405 and that all the batches in the core are assumed to have the same volume, it is possible to express the radius Xo which will separate the inner zone from the peripheral batch (see Figure 3.2.4) as:

$$Xo = \sqrt{(\pi*(2.405)^2/n) * ((n-1)/\pi)} \quad \text{or}$$

$$Xo = 2.405 \sqrt{(n-1)/n}$$
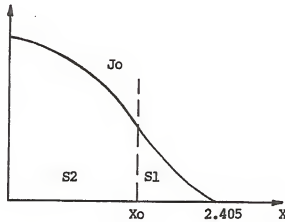
Figure 3.2.4. Bessel-shaped Flux Distribution.

It is now necessary to obtain the ratio of S1 to S1+S2. Recalling that the integral of the Jo Bessel function can be obtained as

$$\int_o^X 2 * \pi * X * Jo(X) * dX = 2 * \pi * X * J_1(X)$$

it is then possible to express the ratio $S1/(S1+S2)$ as

$$S1/(S1+S2) = \frac{2.405 * J_1(2.405) - X_0 * J_1(X_0)}{(2.405 * J_1(2.405))} \text{ or}$$

$$S1/(S1+S2) = (1 - (n-1)/n) * \frac{J_1(2.405 * (n-1)/n)}{J_1(2.405)}$$

where the J_1 Bessel function can be expressed in a series development whose first three terms are

$$J_1(X) = (X/2) - (X/16) + (X/384)$$

This completes the calculation of $S1/(S1+S2)$. If the whole shape factor is now included in the peripheral batch non-leakage probability calculation, the expression becomes

$$Pnlb = 1 - \frac{((1-Pnlr) * \Sigma_i \bar{\phi}_i / \bar{\phi}_b) * (\phi_b^i / \Sigma_i \phi_i)}{(S1/(S1+S2))} * Pc$$

where $\bar{\phi}_i$ are the batch fluxes of the ideal reactor (following Bessel's J_0) and ϕ_b^i are the actual batch fluxes, obtained from the fuel burning calculations. Pc is the proportionality constant or scaling factor, which value was adjusted for the present system to 0.35 and it can be reevaluated for different core configurations using simple diffusion theory codes. Note that in the expression above, $S1/(S1+S2)$ is equivalent to $\bar{\phi}_b / \Sigma_i \bar{\phi}_i$ since these were the fluxes associated with the equivalent homogeneous bare reactor.

Simplifying assumptions are made at this point in order to reduce data input to CRIBUR. The assumptions and approximations

made below are based on the observation of the results of the neutron transport—theory pin-cell depletion code EPRI-CELL.

a). τ, which depends mainly on the core's fast diffusion coefficient, is assumed constant since the core fast diffusion coefficient experiences less than a 2% change through life. The value taken for τ was 40 cm.

b). The geometric buckling is calculated as that of an infinite cylinder whose radius is the radius of the reactor plus reflector savings. As explained before, the reactor is considered infinite in the axial direction because leakage in that direction affects all batches in the same proportion, and this effect is taken care of in the assembly burnup PDQ-7 calculation. The radial leakage, however, affects only the peripheral batch, and is accounted for in the core modeling.

c). For the calculation of the reflector savings, the reflector is assumed to be "thick", since the reflector thickness is 34 cm, which is many times larger than the reflector diffusion length. The savings is then obtained as

$$\delta = Dc * Lr / Dr$$

where Dc = diffusion coefficient of the core.

Lr = diffusion length of the reflector.

Dr = diffusion coefficient of the reflector.

d). Lr and Dr are assumed as a function of the soluble boron concentration of the water. Since Lr can be expressed as

$$Lr = \sqrt{Dr / \sum ar}$$

where ar = macroscopic absorption coefficient of the reflector,

it is first necessary to obtain a parametric expression of Dr and $\sum$ar in terms of the soluble boron concentration. Observation of the results of the pin cell code (EPRI-CELL) shows that both variables can be closely expressed as a linear function of the soluble boron concentration of the coolant, with a regression coefficient very close to 1. The regression lines obtained for Dr and ar are

$$\sum ar \ (cm-1) = 0.0166 + (2.727 \ E-5) * ppm.$$
$$Dr \ (cm) = 0.4033 - (5.135 \ E-6) * ppm.$$

where ppm = concentration of soluble boron in parts per million.

e). The variables Dc (used in the expression above) and L of the core (needed for the calculation of the non-leakage probability during the thermal diffusion period) are also approximated with a straight line. For any particular cycle of core fuel residence, the approximations are accurate to a few percent, but the regression lines change from the first to the second cycle of core residence of the fuel. Since the present study considers an out-in-in fuel management scheme, the fuel in the periphery is in the first cycle of core residence, and since the peripheral fuel

batch is the one dominating the core radial leakage, the straight fits of Dc and Σac are made for the first residence cycle. Should another fuel management scheme be used (e.g. an in-out-in, for a low leakage core configuration), the regression lines would need to be changed, obtaining them from the pin-cell burnup code calculations. For the present case the regression lines obtained for these variables are:

$$\text{Dc (cm)} = 0.4860 - 1.90 \text{ E-7} * \text{BU}$$
$$\Sigma \text{ ac (cm-1)} = 0.1912 + 6.42 \text{ E-7} * \text{BU}$$

where BU = average burnup of the peripheral batch, in MWd/MTU.

Once the peripheral batch has its infinite multiplication factor modified with the probability of non-leakage, it is possible to proceed to the calculation of the core effective multiplication factor. At this point, CRIBUR provides for one additional consideration by allowing any batch in the core to contain burnable poisons. The burnable poisons are handled as a reduction of the batch's infinite multiplication factor by a value that varies linearly as a function of burnup between the user-provided values for BOL and EOL. If no batch is defined as containing burnable poisons, no further variations of batches' multiplication factors are made. In the case where the burnable poison option is to be used and the batch-wise worth of the poisons at BOL and EOL are not known, it is always possible to compute them with the codes involved in the normal burnup calculation. Specifically, EPRI-CELL is able to provide cross sections for lumped burnable poisons (LBP's) to be used in PDQ-7 at BOL and EOL (25). If then PDQ-7

quarter-assembly eigenvalue cases are run with and without the LBP's at BOL and EOL, it is possible to obtain the reactivity worth of the poisons for use in CRIBUR.

CRIBUR obtains the core effective multiplication factor as

$$Kc = 1 \ / \ (\ \Sigma_i (Fi/Ki))$$

where Fi = fraction of the core power produced by the "i"-th batch.

Ki = infinite multiplication factor of batch "i" , modified for leakage and/or LBP's, if any.

The Ki's are obtained from the quarter-assembly PDQ-7 calculations, with the possible modifications described above for any possible radial leakage and/or burnable poison presence. The Fi's are calculated as follows:

$$Fi = \frac{(Ki \ * \ Pi)^2}{\Sigma_i (Ki \ * \ Pi)^2}$$

where Pi = non-leakage probability for batch "i".

The above formula is purely an empirical formula which tries to assign a fraction of the core power to each batch making it as close as possible to the real power sharing encountered in an actual core. This formula evolved from the expression

$$Fi = Ki^2 \ / \ \Sigma_i \ Ki^2$$

used by M.I.T.'s research team (15), but was modified by the author to the expression shown before which adjusts better to the batch

power sharing encountered in a PWR core with an out-in-in in-core fuel management scheme.

With the preceding formulas, CRIBUR takes the burnup data provided by a PDQ-7 1/4 assembly fuel-burnup calculation with a structure similar to the one shown on Figure 3.1.6. and computes the criticality status of the core at the different core-life timesteps, as represented by the graph at the bottom of the figure.

If it is determined that the core is critical at all points in its life, the PDQ-7 1/4 assembly burnup calculation is really representative of the core life, in the sense that the core cycle length, the batch power level and the soluble boron concentrations are consistent. At that point, the PDQ-7 data and the complementary CRIBUR data that will be discussed later give the description and main characteristics of the core burnup (discharge burnup, batch burnups, core life, isotopics, etc.). However, in a general case (and mainly if the PDQ-7 burnup calculation was assigned first-guess values) it is most probable that the core does not maintain criticality through all its cycle timesteps. The first action to be taken is to adjust the soluble boron letdown curve, forcing the core to be critical at all times. This action requires previous knowledge of the soluble boron reactivity worth, and by altering the boron letdown curve, the core life estimate is altered (since it is estimated as the time when soluble boron concentration must become zero). This will force an adjustment of core parameters, as will be explained later.

The way in which CRIBUR accounts for the reactivity worth of soluble boron is also a unique feature in a core model of this simplicity. To the best of the author's knowledge, any code of a complexity level comparable to CRIBUR's considers soluble boron worth as an empirical constant, hardwired in the code; frequently this variable is not established or used, as discussed in Section 3.1. However, an analysis of EPRI-CELL calculations show that soluble boron worth may vary as much as 25% in the range of concentrations that may be normally encountered in a real PWR. In addition to that, soluble boron worth was shown to depend heavily on the fuel cell geometry (see Chapter 2). This suggested the need to evaluate soluble boron worth for each particular reactor case, and if at all possible, it should be evaluated for different concentration levels.

CRIBUR will accept a soluble boron worth constant if so indicated by the operator, or may even provide a default constant which is representative of the average value found in a standard PWR. However, it also provides the opportunity of calculating it for each case and to make it a function of the boron concentration.

If a soluble boron worth calculation is desired, it is necessary to specify some "fake-burnup" steps in one of the fuel-burnup cycles, in a similar way as explained on Figure 3.1.5. Since the fuel-burnup calculation performed to provide data to CRIBUR follows the soluble boron letdown curve for each core cycle through he whole fuel life, the calculation has some burnup steps in which the boron concentration is high, some in which it is near

average, and some in which it is low (see Figure 3.2.5). It is
then possible to specify some "fake-burnup" steps in which soluble
boron concentration is altered by about 100 ppm. This causes a new
eigenvalue calculation, but does not affect the spectrum for the
real burnup step, which is performed with the adequate boron
concentration. By repeating the "fake-burnup" step procedure at
several boron concentration levels, CRIBUR can obtain the soluble
boron worth per ppm at the different concentrations. Observation
of different cases showed that the variation of soluble boron worth
is a closely linear function of boron concentration. CRIBUR
calculates then the regression line and obtains an expression that
allows the calculation of soluble boron worth for any
concentration.
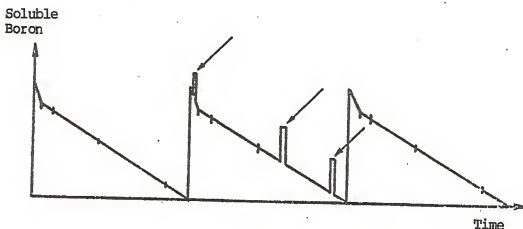
Figure 3.2.5. Fake-burnup Steps for Soluble Boron Worth.

Obviously, if the regression line is calculated for each of
the core cycles that form the total fuel life, slightly different

lines are obtained because of the varying isotopic compositions of the fuel. It is then advisable to perform the linear fit calculation on a cycle towards the middle of the fuel life (the second cycle in the case of a three-batch core) in order to obtain a soluble boron worth regression line that can be representative of the whole core.

One may be concerned about the fact that by defining the "fake-burnup" steps for the soluble boron worth calculation, there is one cycle in which the burnup timesteps do not match with the other cycles. CRIBUR solves the problem by deleting the "fake-burnup" timesteps after the regression line has been calculated. This causes no misadjustment as far as matching timesteps from each cycle, because the "fake-burnup" steps were insignificant in terms of both burnup and time.

Once the soluble boron worth can be obtained for any boron concentration, it is a simple matter to adjust the boron letdown curve so that the reactor is critical for all timesteps in the cycle. However, this causes a major problem in the core life calculation, as discussed below for a reactor with a soluble boron letdown curve such as the solid line of the top of Figure 3.2.6. After a criticality calculation by CRIBUR, it is found that the letdown curve was somewhat in error, and it should be more like the one represented in a dashed line. This new letdown curve was calculated by using the burnup timesteps marked on the old letdown curve, and adjusting the boron concentration for criticality. But if this new curve were indeed to be the actual letdown curve, the
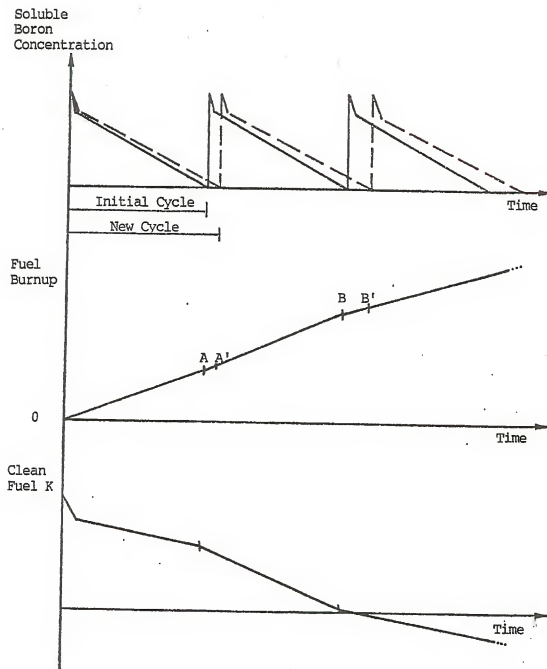
Figure 3.2.6. Problem of Erroneous Cycle Length.

beginning of the second and third cycles would not be at A and B, but rather at A' and B'. which have a different burnup level. The boron concentration that was obtained for criticality by combining points 0. A and B must be obviously excessive. since the "new" points should be 0. A'. and B'. where A' and B' have a larger burnup. and therefore lower reactivity. In short. this means that when calculating a new boron letdown curve which results in a new cycle length. the new curve is automatically in error. because it was not calculated with burnup timesteps whose burnup levels were consistent with the cycle length assumed by the curve.

To solve this problem. one might be tempted to run a PDQ-7 burnup calculation with the new cycle length estimate, but this would imply large number of PDQ-7 iterations and data handling back and forth. destroying one of the main goals of this calculatonal scheme: reduced calculational costs and data handling. To make any further adjustments to the cycle length and the letdown curve with the data available to CRIBUR. it is necessary to find some value or some characteristic that would be implicit in the data and that would not depend on the cycle length. Such characteristic may be found if an approximation is accepted by assuming that the initial guesses of the cycle length and the soluble boron letdown curve are not in large error. If this is not the case. the cycle variation at the end of the calculation will be so large that it will force a new PDQ-7 calculation. anyway. If those guesses are not too erroneous. then. the fuel was burned following a spectrum which is correct within the desired accuracy. and therefore. its isotopics

along the life are within an acceptable margin of accuracy. If the soluble boron is not considered for a moment. the fuel would have a multiplication factor which would depend on its isotopics. which in turn are closely related to the burnup level since the actual burnup was calculated with a reasonable neutron spectrum. This means that under those conditions. the fuel has an invariable characteristic: the correspondence of its "clean" (boron-free) multiplication factor with the burnup level. If this "clean" multiplication factor can be obtained for each known burnup level. it would be a set of data independent of the variations of cycle length (again. as long as the cycle length guess was not too much in error to start with). This is represented in the lower axis of Figure 3.2.6.

Since CRIBUR calculates the soluble boron worth line. it is possible to integrate the reactivity variation caused by boron from concentration zero to the actual concentration assigned to each burnup timestep, and therefore it is possible to calculate the "clean" multiplication factor that can be associated with each initial timestep. and what is more important. that can be associated with each timestep's burnup level. This correspondence is kept in a table within CRIBUR. and for any burnup level, the "clean" multiplication factor can be interpolated. The actual multiplication factor of any particular situation is then computed by first obtaining the "clean" multiplication factor. and then subtracting the integrated soluble boron worth from zero boron to the boron level that is actually assigned.

With this new system. it is no problem if the newly calculated boron letdown curve implies a different cycle length: the last timestep of each cycle is adjusted in length in order to fit the new cycle life. new burnup levels are calculated for each timestep, and CRIBUR proceeds to a new criticality calculation which will yield a new boron letdown curve and a new cycle length. The iterations are continued until the point where the burnup levels. the cycle length and the soluble boron letdown curve are consistent within a certain margin.

As it can be realized by observation of the expressions used by CRIBUR for the criticality calculations. this iterative process involves a simultaneous adjustment of the relative power shared by each batch in the core.

Once the CRIBUR calculation has converged internally, CRIBUR outputs the new estimates of the core life. the soluble boron letdown curve. the relative powers assigned to each timestep. the batch burnup levels. and the discharge burnup level. All these data have then to be compared by the user to the data that were specified for the PDQ-7 fuel burnup calculation. and if any significant differences are encountered (above a certain error margin dictated by the requirements of the study) the PDQ-7 fuel burnup calculation must be repeated with the fuel life data indicated by CRIBUR's output. Figure 3.2.7 shows a flowchart of the calculations performed by CRIBUR which may further clarify its operation and its two main loops: the criticality and cycle life calculations, and the adjustment of cycle timesteps' burnup levels.

```
                              ┌─────────┐
                              │  START  │
                              └─────────┘
                                   │
                          ┌────────────────┐
                          │  Input Option  │
                          └────────────────┘
                    ┌──────────────┴──────────────┐
          ┌──────────────────┐          ┌──────────────────────┐
          │   Data Input     │          │   Data Input From    │
          │  From Keyboard   │          │  Pre-saved Disk File │
          └──────────────────┘          └──────────────────────┘
                    └──────────────┬──────────────┘
      ┌──────────────────────────────────────────────────────────┐
      │  Data Input and Array Dimensioning According to Problem Size │
      └──────────────────────────────────────────────────────────┘
                                   │
          ┌──────────────────────────────────────────────┐
          │   Optional Modification of Problem Data       │
          │      - For Error Correction                   │
          │      - For Modification of Disk File          │
          └──────────────────────────────────────────────┘
                                   │
          ┌──────────────────────────────────────────────┐
          │   Optional Storage of Case Data in Disk File  │
          └──────────────────────────────────────────────┘
                                   │
      ┌──────────────────────────────────────────────────────────┐
      │  Calculation of Step-wise and Cycle-wise Burnup Evolution  │
      │  of Fuel According to Assembly Burnup Data (Input Data)    │
      └──────────────────────────────────────────────────────────┘
                                   │
      ┌──────────────────────────────────────────────────────────┐
      │  Initialization of Work-core Matrix with K's, Time Lengths,│
      │  Burnups, Boron Concentrations and Relative Powers         │
      └──────────────────────────────────────────────────────────┘
                                   │
  ┌──────────────────────────────────────────────────────────────────┐
  │  Calculation of Regression Line of Boron Worth vs. Boron ppm from: │
  │      - Timesteps Specified for such Purpose                        │
  │      - User-defined Constant                                       │
  │      - Default Value                                               │
  └──────────────────────────────────────────────────────────────────┘
                                   │
  ┌──────────────────────────────────────────────────────────────────┐
  │  Elimination of Fake-burnup steps Used for Boron Worth Evaluation  │
  │  Restructuration of Work-core Matrix and Pointers                  │
  └──────────────────────────────────────────────────────────────────┘
                                   │
      ┌──────────────────────────────────────────────────────────┐
      │   Calculation of Table of Unborated Steps K's vs. Burnup   │
      └──────────────────────────────────────────────────────────┘
                                   │
                                  (A)◄──────────────────────────┐
                                   │
  ┌──────────────────────────────────────────────────────────────────┐
  │  Calculation of "Present Core Status" Step-wise and Cycle-wise Burnups │
  └──────────────────────────────────────────────────────────────────┘
                                   │
                                  (B)◄──────────────────────────┐
                                   │
      ┌──────────────────────────────────────────────────────────┐
      │  Interpolation of "Present Core Status" Clean K's for all Steps │
      └──────────────────────────────────────────────────────────┘
                                   │
      ┌──────────────────────────────────────────────────────────┐
      │  Calculation of K's from Current Burnup and Assigned Boron ppm │
      └──────────────────────────────────────────────────────────┘
                                   │
      ┌──────────────────────────────────────────────────────────┐
      │   Modify Batch K's for Burnable Poisons Presence (if any)  │
      └──────────────────────────────────────────────────────────┘
                                   │
      ┌──────────────────────────────────────────────────────────┐
      │  Calculation of Escape Probabilities for Peripheral Batch  │
      └──────────────────────────────────────────────────────────┘
                                   │
          ┌──────────────────────────────────────────────┐
          │  Modification of Peripheral Batch K for Leakage │
          └──────────────────────────────────────────────┘
                                   │
                            ┌─────────────┐
                            │  continued  │
                            └─────────────┘
```

Figure 3.2.7. Flowchart of CRIBUR Code

```
                              ↓
                      ┌──────────────┐
                      │ continuation │
                      └──────────────┘
                              ↓
            ┌──────────────────────────────────────┐
            │ Calculation of Steps' Relative Powers │
            └──────────────────────────────────────┘
                              ↓
          ┌─────────────────────────────────────────────┐
          │ Calculation of Core Effective Multiplication Factor │
          └─────────────────────────────────────────────┘
                              ↓
        ┌───────────────────────────────────────────────────┐
        │ Calculation of New Boron Concentrations for Criticality │
        └───────────────────────────────────────────────────┘
                              ↓
        ┌─────────────────────────────────────────────────┐
        │ Calculation of New Core Life and New Timestep Lengths │                 ↑
        └─────────────────────────────────────────────────┘
                              ↓
              ┌─────────────────────────────┐
              │ Apply Convergence Accelerator │
              └─────────────────────────────┘
                              ↓
┌───────────────────────────────────────────────────┐              ┌────────┐
│ Core K's Close Enough to 1 Before Boron Adjustment ? │── NO ──│ Go to A │──┘
└───────────────────────────────────────────────────┘              └────────┘
                             YES
                              ↓
      ┌───────────────────────────────────────────────────────┐
      │ Calculation of Step-wise Burnups With New Set of Relative Powers │
      └───────────────────────────────────────────────────────┘
                              ↓
        ┌──────────────────────────────────────────┐
        │ New Cycle Burnups Agree With Old Iteration's ? │──── YES ──┐
        └──────────────────────────────────────────┘           │
                             NO                                  │
                              ↓                                  │
          ┌───────────────────────────────────────────────┐     │
          │ Apply Convergence Accelerator for Relative Powers │     │
          └───────────────────────────────────────────────┘     │
                              ↓                                  │
            ┌────────────────────────────────────────────┐       │
            │ Calculation of "Present Core Status" Burnups │       │
            │ Storage as "Old Iteration" Burnups          │   ↑   │
            └────────────────────────────────────────────┘       │
                              ↓                                  │
                       ┌──────────┐                              │
                       │ Go to B  │──────────────────────────────┘
                       └──────────┘
                              ↓
                    ┌───────────────┐
                    │ Print Results │◄──────────────────────────────
                    └───────────────┘
                              ↓
                        ┌─────┐
                        │ END │
                        └─────┘
```

Figure 3.2.7. Continuation.

3.3. Benchmarking of the Burnup Calculations

3.3.1. Available References for Comparison.

In order to benchmark the burnup calculational scheme developed for the calculations performed in this study, it is necessary to have a quality assessment for each step in the chain of programs which compose the total calculational scheme. The programs used in performing the point burnup calculations are industry standards. They already have had sufficient benchmarking (26,27) as to make additional work in this direction by the author unnecessary. However, it is still necessary to check the behavior of the core model against other models of well-known characteristics, or if possible, against an actual reactor's behavior.

It is clear that the best benchmarking would be the comparison with a true reactor behavior in an equilibrium cycle, since this is the situation that the core model tries to represent. In order to use such a benchmark it would be necessary to know all the exact compositions of the equilibrium-cycle real core. Unfortunately, in an equilibrium-cycle core, the compositions of the once-burned and twice-burned batches are not exactly known from an experimental basis, and the analytical results are not published in the open literature, rendering this benchmarking possibility impractical.

There is, however, one case in which the model can be compared to a real-life reactor: at B.O.L. of a first core. In such a situation, all batches in the core are new and clean, and thus their compositions are known, as well as the soluble boron

concentration that holds the core critical at hot, full power, and all control rods out (HFP, ARO) conditions. It is then possible to calculate the multiplication characteristics of each separate batch, and use them as an input to CRIBUR. Obviously, this method allows the benchmark of the criticality calculation part of CRIBUR, but not of the power distribution and burnup assignment part.

Another way of benchmarking CRIBUR is, as said above, to compare its results against those of some well-known and accepted codes. The code PDQ-7 is a good candidate, because it has been widely accepted as one of the main calculational tools in the nuclear industry, its accuracy is well known, it can explicitly represent the geometry of a 1/4 core, and it is possible to input in it exactly the same data that is needed in a CRIBUR case, thus making it possible to have exactly the same case represented in both models.

However, this 1/4 core PDQ-7 representation has a few problems. In a real equilibrium core, the assemblies composing any fuel batch have different characteristics depending on their particular history, and there are differences even between the pins that form a fuel assembly, because some of the pins are at the assembly periphery, facing assemblies from other batches (and therefore with sharply different characteristics and history) while other pins lie inside the assembly, surrounded by pins of their same batch and very similar life conditions. These differences cause the interfaces between batches and between different assemblies to be "smoothed out" as burnup increases, naturally

limiting local power peaking effects. Unfortunately, if a PDQ-7
1/4 core model is prepared with data identical to that used in a
CRIBUR run, each batch has one single composition which is assigned
to all of its pins, and no differences between pins with their
"smoothing" effect on power distributions exist. The compositional
differences between neighboring pins from different batches are
sharper than they would really be in an actual core, causing
exagerated effects on the power peakings. This problem invalidates
the PDQ-7 1/4 core model for the purposes of benchmarking power
distributions and burnups, although it can still be used as a
redundant reference for benchmarking of the criticality
calculations of CRIBUR, since the overall core multiplication
factor is not affected by the somewhat inexact power peaking
analysis.

A PDQ-7 run for a 1/4 core configuration, even used without
PDQ's burnup features (i.e. just for criticality and power
distribution "snapshot" analysis) presents the additional
difficulty of its involved input for a complex geometry that
includes several different composition zones. This is primarily a
problem of skill and time, and can still be reasonably viewed as a
method for benchmarking purposes.

The benchmarking references described so far allow for a good
checking of the criticality calculation part of the code, since
both the PDQ-7 1/4 core batch-homogenized model and the
real-reactor B.O.L., H.F.P., A.R.O. give good values of the
multiplication factor, and it is easy to obtain the necessary core

data to reproduce the situation on CRIBUR. However, it is still necessary to obtain a reasonable source for comparison of the power distribution among the different batches in CRIBUR. Since CRIBUR analyzes an equilibrium-cycle core, it was considered that the best method would be to compare its power distributions at different points in core life with those appearing in real cores working in an equilibrium cycle (say cyles 3 or beyond). The actual core compositions would not be known, but the power distributions on a batch-wise fashion are likely to be similar, and provided the cores taken for comparison are similar to the one described in CRIBUR, the code's estimates should be very close to the values seen on the real reactors.

3.3.2. Criticality Benchmarking.

As described in the previous section, two different references were used for the benchmarking of CRIBUR's criticality calculations. One reference was composed of several non-burnup runs of PDQ-7 with a 1/4 core configuration, where each assembly was explicitly represented, although the geometry mesh used was somewhat coarser than an explicit pin-by-pin representation. Each assembly was given a uniform composition equal to the one existing in the corresponding batch in the CRIBUR calculation. Macroscopic cross sections were used for each region instead of microscopic, and this simplified data handling. There was no need for microscopic cross sections, since no isotopic changes were allowed. Two energy groups were used, as recommended by EPRI for PWR calculations with the ARMP code package.

The PDQ-7 1/4 core criticality tests were performed at four different points in the life of an equilibrium core: at B.O.L., at approximately 1/3 of core life, at about 2/3 of core life, and around E.O.L. . In all the cases tested, the effective multiplication factor calculated by CRIBUR differred from PDQ's by less than 1%, and in two out of the four cases the difference was under 0.3% .

In an attempt to check the importance of some of the calculational features included in CRIBUR and not normally present in other simple burnup schemes, the same four cases were recalculated in a version of CRIBUR in which the calculation of the leakage probability for the outer batch was bypassed. In all four cases the error of the multiplication factor as compared to PDQ's was over 4%, having increased from the original CRIBUR value by about 3.5 to 4%.

Another check of the accuracy achieved by CRIBUR as compared to other simple schemes was done by calculating the core effective multiplication factor (K-eff.) as an average of the batches' infinite multiplication factors (K-inf.'s), as indicated by Graves(28). This K-inf. averaging method yielded errors between 2.6 and 3.7% with respect to PDQ's results, which represented errors about 2.5% higher than the ones incurred by CRIBUR in the representation of identical cases. Table 3.3.1 shows the values obtained for all the benchmark and comparative cases described above.

Table 3.3.1. CRIBUR and Batch-averaging Method Criticality Errors.

Cycle Status	CRIBUR K-eff.	% Diff to PDQ	No-leak CRIB. K-eff.	% DIFF to PDQ	BATCH-AVG K-eff.	% DIFF. with PDQ
B.O.C.	.9694	.23	1.0094	4.36	0.9930	2.66
1/3 cycle	.9520	.15	0.9909	4.23	0.9772	2.79
2/3 cycle	.9274	.68	0.9617	4.40	0.9524	3.38
E.O.C.	.8996	.98	0.9304	4.44	0.9235	3.67

The other reference used for benchmarking of CRIBUR's criticality calculations was a real reactor at B.O.L. for the first core. The reactor chosen was Comanche peak, Unit 1(27). The reasons for choosing this particular core were the following:

a). The reactor was of the same 4-loop Westinghouse type with 17X17 pin fuel assembly as the one used for the CRIBUR calculations. This made all dimensions, compositions and temperatures common for both the real reactor and the data used for CRIBUR's calculation.

b). Being at B.O.L. of the first core, the compositions of all batches were known, since no burnup had occurred yet, and therefore no fission products existed in the core.

c). The report on the real reactor indicated the HFP, ARO soluble boron concentration needed for criticality at that time, as well as the worth of the burnable poison rods at B.O.L.. all of which made the comparison of the reactor with CRIBUR a rather straight forward procedure.

The comparison of CRIBUR with Comanche Peak 1 B.O.L. yielded a K-eff. of 1.013 after accounting for the reactivity worth of the burnable poisons. CRIBUR had, thus, a 1.3% K-eff. error with respect to the real reactor. It may be of interest to note that B.O.L. of the first cycle is probably the worst moment for CRIBUR to give a good estimate of the core criticality, due to the sharp interfaces existing in the core because of the different types of fuel and poisons present. However, even in this situation, CRIBUR's criticality calculation was acceptably accurate.

A comparison of criticality evaluations of the CRIBUR scheme and the scheme used by M.I.T. (15) whose main formulas were outlined in Section 3.2 shows differences between 1.6% and 2.3% for the core multiplication factor. The main errors of the M.I.T. scheme appear on the batch power distribution, as will be presented later.

Some runs were made with CRIBUR in which some of the factors of the calculational scheme were altered, in order to observe the sensitivity of the results to the presence or accuracy of those different factors. In a test where the Shape Factor was given a constant value of unity (therefore implying that for leakage purposes the flux shape followed a Bessel function) the core multiplication factor experienced a sharp increase of 2.76% at EOL and 3.26% at BOL. It is logical that these forced lower-than-real leakages caused a larger effect at BOL, when the external batch, which is experiencing the neutron leakage, is most reactive.

In a series of tests where the batch multiplication factors were altered by 1%, the core multiplication increased around .3% when the altered batch was the oldest one, about .36% when the second batch was altered, and approximately .32% when the alteration affected the first batch. The sensitivity of the core to the accuracy of the batches multiplication factors depends on the batch power sharing. The code showed some sensitivity to the neutronic characteristics of the reflector: if the reflector was considered absent of soluble boron for the radial leakage calculation, the core multiplication factor showed a variation of .11% at BOL (when the real boron concentration is highest) and .04% near EOL, when the level of soluble boron is low. This demonstrates that the variations of the reflecting characteristics of the reflector were non-negligible.

The percentage variations of core multiplication factors associated with the different altered factors for five points during the core life are shown on Table 3.3.2.

Table 3.3.2. Effects of Code and Data Alterations on Core K-eff.

Core Life Time:	B.O.L.		M.O.L.		E.O.L.
Alteration	Percentage Change on CRIBUR K-eff.				
Shape Factor = 1	3.26	3.01	2.99	2.94	2.96
+1% K, Batch 1	0.33	0.32	0.32	0.32	0.31
+1% K, Batch 2	0.36	0.37	0.36	0.36	0.36
+1% K, Batch 3	0.29	0.30	0.30	0.30	0.26
Unborated Reflec.	0.11	0.10	0.09	0.08	0.04

3.3.3. Power Distribution Benchmarking.

Since it is not possible to obtain actual accurate equilibrium core compositions for the type of reactor modeled in this study (Westinghouse's 4-loop, 17X17 pin assembly) because there is not yet any such reactor in an equilibrium cycle, the benchmarking of the batch-wise power distributions calculated by CRIBUR was done by comparing its results for the standard case with the batch-wise power distributions of several Westinghouse cores in their equilibrium cycles.

Typical power distributions obtained by CRIBUR yield relative powers of 1.02, 1.09 and 0.88 for the three batches in the core, from newest to oldest respectively, with no burnable poisons present in either of them.

The power distributions observed in some real cores in their equilibrium cycles are (from newest to oldest batch respectively): 0.99, 1.05, 0.94 for Zion II ; 0.96, 1.13, 0.9 for D.C. Cook ; and 0.97, 1.15, 0.88 for H.B. Robinson 2. Core-following calculations performed by utilities show assembly-wise power level errors of a few percent, reaching sometimes 10%. EPRI's procedures manuals list as acceptable an error of 5%.

Obviously, when the partial power of all the assemblies are added up into batches, the calculations come closer to the real values, but this is achieved with pin-by-pin or nodal multi-dimensional calculations of a degree of sophistication far superior to CRIBUR's. The results obtained by CRIBUR tend to assign a few percent more power than real to the fresh batch, and a

few percent less to the second batch. What is also important is that the order of batches from the highest to the lowest power level is always kept the same as in the real cores in spite of the closeness of the power levels of batches 1 and 2. The calculational scheme used by M.I.T. yields a too high power level for the first batch, while reducing that of the second batch. The variations are of such magnitude that the second batch does not hold the maximum power as is the case in an actual core, and as is the case with CRIBUR results, but the maximum power sharing is assigned to the first batch. It is important to remember that CRIBUR's calculations did not assign any poisons to the fresh batch, which are some times used in actual reactors.

From these comparisons it is possible to conclude that CRIBUR's estimates of batch-wise power distributions are very close to the true values found in an equilibrium-cycle PWR and quite acceptable for relative burnup studies.

The alterations mentioned in the criticality benchmarking section affected the power distribution, the discharge burnup and the cycle length, proving again they are non-negligible in nature. The elimination of the Shape Factor in calculating the radial leakage boosts the peripheral batch power and reduces the power sharing of the second and third batches. The discharge burnup shows a large erroneous increase of 6.3%. This is due to the core experiencing a very reduced radial leakage.

The total elimination of the radial leakage shows similar effects, but in a yet larger scale. The discharge burnup is increased 7% over the correct calculation.

Increasing the batches' multiplication factors by 1% increases by a few percent the power sharing of the batch being altered, and causes an increase in discharge burnup of 2% when the outer batch is altered, and 1.7% when one of the internal batches is altered.

The sensitivity calculations were performed using burnup data corresponding to the standard, full power PWR.

CHAPTER IV
DETAILED SHARP BURNUP OPTIMIZATION STUDIES

4.1. Cases Chosen for Final Study

Once the burnup calculational method is developed and the set of programs to be used are tested (as described in Chapter III), it is necessary to define the burnup cases that have to be considered for this study. It is important to realize that although the burnup calculational scheme developed for this work requires a reduced level of computational effort for a quality calculation, a considerable amount of work is required to complete each burnup study. It is not only necessary to complete the chain of EPRI-CELL, NUPUNCHER, PDQ-7 and CRIBUR, but it normally takes two or three iterations of PDQ-7 and CRIBUR before all the core cycle data converges. This represents a significant volume of data handling, and thus, it is important to avoid burnup calculations that are not required for the purpose of the study.

Since the main concern of this study is to determine the effects of reduced power densities on the neutronics, fuel utilization and general isotopic behavior of the reactor, it is obviously necessary to examine several power density levels. Two different plant arrangements were considered as possible in Chapter I for the low-power cores; the main designs were the multiple-core plant with or without preheater core, and the

single-core plant with reduced power output. As already indicated in Chapter I, one of the key requirements for acceptance of the SHARP concept by the industry is the minimization of significant redesign, and this requirement is only fulfilled by the single-core plant concept. The preheater reactor was also excluded from consideration for the same reason. Only the single-core design is, therefore, considered in the in-depth burnup studies.

The preliminary burnup calculations described in Chapter II use a 25% power level core as the SHARP core; this would correspond to a nuclear plant with only about 300 MWe power output. It is not very likely for a plant of such reduced power output to result economically attractive. This aspect will be further discussed in Chapter V. An adequate power level for a reduced-size plant may start at about 500 MW, which would correspond to about 40% of the power level of a standard plant. However, since studies of cores with power densities around 75% that of the standard core have already been done (2) (involving significant core redesign), the author considered that the scoping studies will explore a power range between 25% and 50% that of the standard reactor.

A power output of 50% that of the standard plant may be very attractive for plants sited near medium-sized consumption centers. Thus, the burnup study of low-power cores between 25% and 50% power level is likely to give a good perspective of the results that may be expected from the SHARP's in the range more likely attractive to industry. The power levels finally chosen for detailed study were 25%, 33% and 50% of the standard reactor power density.

Given a power density, there are a number of parameters that can be allowed to vary and therefore are susceptible to optimization, without changing much the design of the core. Some of these parameters are: lattice pitch, fuel pin diameter, fuel enrichment, burnable poison loading and/or management, consideration of gadolinium-loaded fuel pins, in-core fuel management scheme, etc. Some of these parameters are already being considered in industry studies directed to improve fuel utilization, and their use is discussed in greater detail in Appendix A. It is obvious that the optimization of all these parameters is well beyond the scope of this work. It is then necessary to choose the best set of parameters that can be used for core optimization within the established constraints of these studies.

The main criteria necessary for the selection of the parameters are the reliability and sensitivity of the calculational methods to the parameter(s) and the acceptability of the change by the industry.

One parameter that could be changed or modified is the in-core fuel management scheme. The calculational method developed in Chapter III is able to accept such change, but the alternative mangement schemes (v.g. in-out-in, explained in Appendix A) are still considered problematic in the industry, and they would have little probability of being accepted. Therefore, this study does not consider any variational in-core fuel management scheme, and

the standard out-in-in scheme is used throughout the burnup optimization calculations.

The use of burnable poisons is very common in the industry, but although they can be represented in the calculational method, it would be hard to reliably account for small variations in them, and comparison with a "standard" core would be difficult, since this is normally a particular case-to-case type of parameter. Gadolinium loaded pins are still far from widespread consideration for commercial PWR's, and the calculational schemes available for this work would not be able to account for them properly, so they are also not considered in the studies.

Enrichment is a parameter that is likely to be varied in most studies, and several works have pointed out that slightly higher enrichments may improve ore utilization, aside from extending core cycles, but this forcedly requires the use of some type of burnable poison, and it is therefore not considered here for the reasons mentioned above. However, further research is needed in this direction if suitable benchmark cases become available, since this is a parameter which might result of significant effect, and which should not present large problems of acceptance by either the industry or the regulatory agencies.

A parameter that is simple to vary and that has shown to have a significant effect in the preliminary calculations is the fuel lattice pitch. Its effects on the neutron spectrum cause significant changes on the isotopic evolution of the fuel, and therefore on the discharge burnup levels, the ore utilization, etc.

Variation of this parameter is likely to be accepted without much problem by the industry, mainly because it is not a significant change in the design and operation of the plant, nor does it require special types of calculations for core-follow control. This parameter is therefore chosen for core optimization in the burnup studies of the low-power cores. Since there are sound guidelines on the range of pitches that are acceptable from the preliminary calculations described on Chapter II, the starting pitch for the parametric studies was selected at 1.20 cm and incremented by 0.05 cm for each burnup calculation, until a net reduction of burnup is encountered. The decrease in burnup is likely to occur around a pitch of 1.45 cm according to the results of the preliminary calculations.

If a 1.45 cm pitch shows the expected decrease in burnup (after experiencing a maximum peak in a smaller pitch), this requires a total of 19 burnup cases: six pitch values for each of the three low-power cores (25%, 33%, and 50% power density), plus one case for the standard core, which is only studied at its nominal pitch, as a reference case. Thermal-hydraulic safety considerations prevent the use of the standard power density core with larger pitches, as explained in next chapter.

4.2. Results of Optimization

4.2.1. Neutronics and Isotopic Results.

The burnup studies performed for the different power level cores and the changing pitches generated data sets that require

processing to analyze the key issues and the significant differences between the various cases studied.

The results shown in this section are directly obtained from the PDQ-7 - CRIBUR burnup optimization calculations or from simple manipulations of their output data.

The first set of data which is needed for core optimization analysis is the U-235 and plutonium isotopes number densities at loading and discharge. However, the number densities per se are not a common means of comparing core performances and they are listed in Appendix D1 for reference purposes. The figures are used later in this chapter in order to obtain other commonly used comparative indices. As stated before, all the burnup studies were done under the assumption of once-through fuel management, and thus, reprocessing was not considered. In these studies, the only fissile material loaded into the core was U-235. However, for the sake of completeness and because it will be necessary to reprocess spent fuel to recycle fissile species, some figures concerning these discharged isotopes have been calculated and are presented in this section.

Table 4.2.1 presents the comparison of burnup achievements for the different cores, fissile species accounting, and ore usage. The first two columns in Table 4.2.1 define the core case studied. Note that the two parameters varied through the study are the power level and the fuel cell pitch. The full-power, standard pitch core results are included as the reference case for comparison purposes. The third column shows the effective core cycle duration as

Table 4.2.1. Burnup Achievements, Fissile Isotope Usage and Ore Usage.

Pitch cm	Power %	Cycle F.P.H.	Dis. BU. MWd/MTU	Dis. Fis. MT/MWd (a)	Rec. Fis. % Dis/Ld.	Net Fis. MT/MWd (a)	Pu Dis. MT/MWd (a)	U3O8 MT/MWd (b)
1.20	50	15588	35172	4.195	47.16	4.619	2.751	1.899
1.20	33	23751	35725	4.020	45.90	4.658	2.685	1.870
1.20	25	32058	36166	3.914	45.24	4.657	2.651	1.847
1.25	100(c)	8018	36183	3.637	42.06	4.931	2.465	1.846
1.25	50	16529	37295	3.418	40.73	4.894	2.414	1.791
1.25	33	24837	37359	3.357	40.07	4.941	2.381	1.788
1.25	25	33359	37634	3.287	39.52	4.950	2.354	1.775
1.30	50	16704	37689	3.054	36.77	5.171	2.218	1.772
1.30	33	25329	38099	2.949	35.90	5.187	2.176	1.753
1.30	25	33924	38271	2.897	35.42	5.203	2.154	1.745
1.35	50	16901	38134	2.720	33.14	5.409	2.071	1.752
1.35	33	25526	38397	2.672	32.77	5.402	2.033	1.740
1.35	25	34162	38541	2.628	32.36	5.415	2.013	1.733
1.40	50	16534	37306	2.636	31.43	5.673	1.967	1.791
1.40	33	24994	37595	2.555	30.69	5.691	1.935	1.777
1.40	25	33415	37698	2.514	30.29	5.710	1.917	1.772

(a):* 1.0 E-7   (b):* 1.0 E-4   (c): Reference Case.

obtained by CRIBUR calculations. The figures are, thus, full-power hours of operation before E.O.L. is reached. Of course, in this context "full power" refers to the total power level assigned to each particular core. Note that for all three reduced power levels studied, the maximum cycle length is obtained for a pitch of 1.35 cm as compared to the 1.25 cm of the standard core. It seems reasonable to assume that the standard power core would also present a longer cycle life with a pitch of 1.35 cm, but thermal-hydraulics and safety limitations prevent the use of such pitch, as explained in Chapter V. The improved burnups achievable with the different pitches for the different fractions of full power translate into cycle lengths higher than the factor of two-, three- or four-fold that could be expected from the cores operating at the 50, 33 or 25 percent power level with respect to the standard core. Thus, the 50%, 33% and 25% power cores with the 1.35 cm pitch show a cycle length which is respectively 2.11, 3.18 and 4.26 times the cycle length of the standard core. This means that for the 50% power core (which is the most commercially attractive of the cases considered) the same initial fuel load would provide not only the natural double core life as compared to the standard plant but an additional 11%. The 33% and the 25% power cores would provide additional cycle lengths of 18% and 26% respectively. However, this additional cycle lengths, although very important from the plant cycle and refueling outage schedule viewpoints, should not be confused with the additional burnup achieved by each core, which is discussed below. The basic

difference is that the burnup level is an absolute measure, independent of the core power level, while the cycle length (even the full-power cycle length, as represented in this third column) is a measure that is dependent on the power level of the plant.

The fourth column of table 4.2.1, labeled Dis. BU. MWd/MTU, shows the burnup levels achieved by each core, as calculated by CRIBUR. This is probably the best measure of the amount of energy provided by a given amount of initial fuel loaded in the different cores for the case when no reprocessing is being done. This measure of burnup does not take credit for discharged fissile materials, which would be of obvious interest in the case of a reprocessing policy. However, under the present U.S. policy, this can not be accounted for, and the burnup achievable is a good measure of the energy obtained from a certain amount of initial fuel. The levels of burnup achieved by the best pitch cell (1.35 cm) for the 50%, 33% and 25% power level cores are 5.4%, 6.1% and 6.5% above that achieved by the standard core, respectively. Referring these burnup increments to a common basis, the 50% power core yields 0.11% extra burnup per percent power reduction with respect to the standard plant; the increments for the 33% and 25% power density cores are 0.092% and 0.087% per percent reduction with respect to the full-power plant. These figures represent a 15% and a 20% lower value than that obtained in the power reduction to 50%. The implicit meaning of these figures is that the additional burnup obtained with low power density cores with respect to the standard plant is largest for the initial power

reductions, but as lower power densities are experienced, the percent increase in burnup per percent power reduction appears to be smaller. This is a significant reason to limit the power reduction to a "safer and optimum level" instead of going to lower power levels, since little is gained in terms of burnup with the further power reductions. Other than the safety-related issue, the improvements in burnup is one of the most significant contributions of the reduced-power cores.

The fifth column of table 4.2.1, labeled Dis. Fis. MT/MWd shows the mass of fissile material that is discharged from each core with the spent fuel. The remaining fissile material is composed of U-235, Pu-239 and Pu-241.

All fresh fuel loaded in the cores consisted of uranium enriched 3.1% in U-235. The total mass of uranium per unit volume loaded with the fresh fuel can be easily obtained by multiplying the number density of U-235 in the fresh fuel and dividing by the enrichment and by Avogadro's number. In a similar manner, the total mass of fissile materials per unit volume present in the discharged fuel is immediately obtained from the discharge number densities of U-235, Pu-239 and Pu-241.

With these, it is possible to obtain a ratio of the total mass of fissile material discharged from the reactor versus the total mass of uranium loaded with the fresh fuel, which is a constant value for all the cores. Recalling that the burnup level of the discharged fuel is expressed in terms of MWd per Metric Ton of uranium loaded, it is possible to divide the above ratio by the

burnup value, which will yield the discharged fissile mass per MWd produced. This is the value displayed in the fifth column of Table 4.2.1.

The calculations show that the total mass of discharged fissile material per MWd of energy produced decreases steadily as the pitch is increased. This is due to the softening of the neutron spectrum caused by the increasing amounts of moderator present in the core, which reduces resonance absorption in U-238, and therefore reduces the production of plutonium isotopes. For each particular pitch, the discharged mass of fissile nuclides per unit energy produced is reduced as the power level of the core is reduced. There are two factors contributing to this effect: first, as the power level is reduced, the Doppler broadening of resonances is also reduced, having the same effect explained above when discussing the production of plutonium isotopes; and second, as the lower-power cores are driven to higher burnups due to the extra reactivities supplied by the lower xenon levels and the reduced Doppler effect, more incremental fissile material is consumed in energy production than is gained during the additional time for plutonium conversion.

These observations about the reduction of discharged fissile nuclides as pitch is increased will have further significance when fuel reprocessing is contemplated, since the further burnup achieved by the fuel would also reduce the worth of the discharged fuel.

The sixth column on table 4.2.1, labeled Rec. Fis. % Dis/Ld shows the percent value of the weight of discharged (i.e. recyclable) fissile nuclides as compared to the weight of loaded fissile material. The values obtained for this column are plotted as the lower set of curves in Figure 4.2.1, for easier interpretation. Since the only fissile material loaded in the core is U-235, this column can be easily obtained by adding the number densities of the discharged fissile species multiplied by their respective atomic weights, and dividing by the same figure, referred to the loaded U-235.

This column shows two main facts: first, it shows more clearly the tendencies observed in the previous column, and second it points out that the mass of discharged fissile materials is between one third and one half of the mass of fissile material initially loaded into the core. This implies the large potential economical value of the discharged fuel when incorporated into a spent fuel reprocessing policy.

The seventh column of the table, labeled Net Fis. MT/MWd indicates the net mass of fissile material spent per MWd thermal produced. In this case, net mass stands for the difference between the fissile mass loaded into the core and the fissile mass discharged with the spent fuel. Subtracting the fissile mass discharged from the reactor from the mass of U-235 loaded with the fresh fuel yields the net usage of fissile mass, and dividing it by the burnup level yields the desired value of net fissile mass used
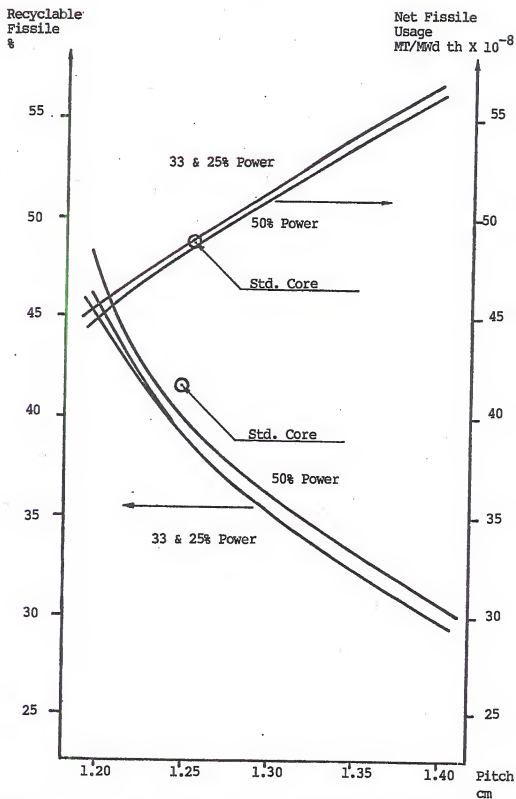
Figure 4.2.1. Recyclable Fissile and Net Fissile Usage vs. Pitch.

per unit energy generated. These values are also plotted in Figure 4.2.1, as the upper set of curves.

For any given pitch, the net fissile consumption shows very small changes (about or less than one percent) as power level is changed, although a slight trend can be seen of increasing net consumption for lower power levels. As pitches are changed, the net fissile consumption increases for more moderated pitches, causing the once-through, burnup optimized pitch of 1.35 cm to consume about 10% more net fissile mass as the present standard pitch. These results are in agreement with the observations on fissile mass discharge appearing on column 5 of this same table. This presents again the controversial point that the 1.35 cm pitch is an optimum point for burnup and ore utilization under the assumption of a once-through fuel cycle. This is obviously not a favorable pitch from the ore conservation point of view if a reprocessing policy is implemented.

The eighth column on Table 4.2.1, labeled PU Dis. MT/MWd shows the total amount of plutonium that is discharged from the core per MWd thermal produced. This figure has obvious interest from the nuclear weapon proliferation point of view. The figures have been obtained with an expression identical to the one used for the discharged fissile mass (shown in the fifth column of the same table) by substituting Pu-240 for the U-235. The plutonium discharge values are plotted in Figure 4.2.2 as the upper set of curves.

From this point of view, the optimized pitch has an advantage over the current standard reactor. For each particular pitch, the amount of plutonium discharged is reduced a few percent as the power level is cut down. This is due to the softer neutron spectrum existing in the lower power cores caused by the reduced Doppler effect and the lower xenon level. As the pitch is increased, the discharge of plutonium is sharply reduced. Thus, the 50% power, 1.35 cm pitch core yields almost 16% less plutonium than the standard reactor for the same energy production. The significant variation in the yield of plutonium associated with the pitch change is obviously due to the spectrum shift caused by the increased amount of moderator present in the larger pitches.

The last column on Table 4.2.1, labeled U3O8 MT/MWd shows the ore use for each of the cores on a per-unit-energy basis, under the assumptions of once-through fuel cycle, uniform enrichment to 3.1% U-235, and a tails enrichment of 0.2% . Using standard tables for feed-to-product ratio in the enrichment cascades (28) the ore use can be obtained with the expression

$$\text{Ore usage (MT/MWd)} = 6.68 \ / \ BU \ \text{(MWd/MTU)}$$

The results for this column are also plotted on Figure 4.2.2, as the lower set of curves. As could be expected, in this case the best burnup cell presents the best ore utilization. Thus, the 50% power core, 1.35 cm pitch uses 5.1% less ore than the current standard reactor, for the same generation of energy. For each particular pitch, the lower power cores show better ore utilization
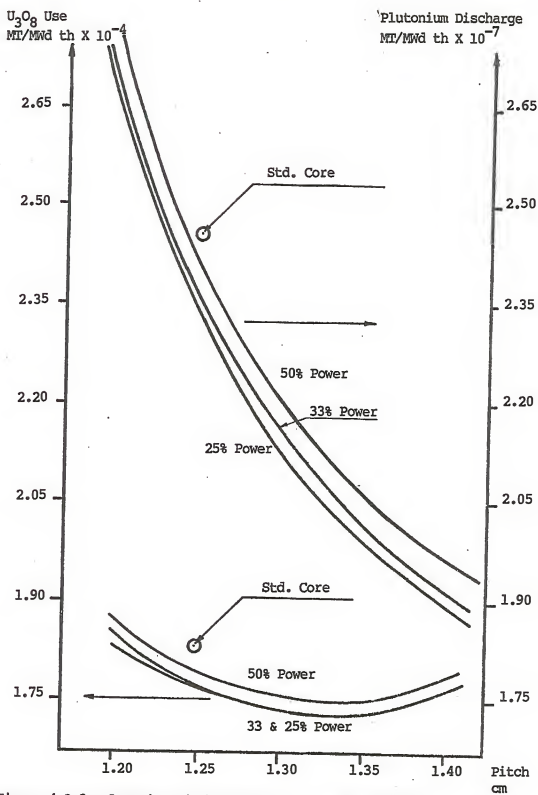
Figure 4.2.2. Plutonium Discharge and U_3O_8 Use. vs. Pitch.

than the higher power ones by some tenths of a percent. This is due to the extra reactivities associated with the reduced Doppler and xenon levels. However, as pitch is varied, ore utilization reaches a minimum at about 1.35 cm pitch, and increases for any further pitch increase or decrease. This results from the existence of two conflicting effects which substantially influence the core cycle length: First, there is the moderating effect of the coolant. Since all LWR's work on an undermoderated configuration for safety reasons, an increase in pitch means additional moderation, and therefore an increase in the multiplication factor of the core. This applies, of course, until the best moderated cell is obtained, which occurs at about 1.5 cm pitch. Second, there is the conversion of U-238 into plutonium. Plutonium accounts for about 50% of the core power at E.O.L., and therefore, the amount of plutonium present in the core heavily affects core life. Since plutonium is converted from the parasitic capture of neutrons in U-238, a better moderated pitch has a softer spectrum and reduces the resonance capture in U-238, resulting in a lower plutonium inventory at E.O.L. (this effect could be observed in the previous column on this table). For very undermoderated pitches, an increase of pitch leads to a larger effect from the additional moderation than from the reduced plutonium yield and core life is increased. When larger pitches are used, the reduction of plutonium inventory outweighs the effect of the slightly softer spectrum, causing a reduction of core life.

4.2.2. Plant Operational Data Variations.

The previous section showed how the variations of power density level and the changes of fuel lattice pitch affect the neutronics characteristics and the isotopic evolution of the core. These variations result in changes in plant operational data which imply sharp differences from both the operations and the economic viewpoints. This section is intended to show the main plant operational characteristics associated with each of the studied cores, and their underlying implications. The data shown in this section are obtained from the burnup and cycle lengths shown in the preceding section, which were based on the PDQ-7 - CRIBUR burnup results.

Table 4.2.2 establishes the basis in which to compare the availability and capacity factors of the different reactors, and their cycle lengths and refueling outages. These parameters are of major importance since they represent large differences from the economical viewpoint. One of the main economical incentives is reducing plant outage time. This is a well known fact, which value increases with time. Presently, depending on the utility location, its dependence on oil, coal, gas, the time of the year, etc., one outage day represents an additional expense of one to two million dollars.

As on Table 4.2.1, the first two columns of Table 4.2.2 are used for description of the core case studied, and refer to the power level and the pin array pitch. The columns that follow on

Table 4.2.2. Plant Operational Data Comparison.

Pitch cm	Power %	Real Cycle Days	Life Reful Days	Refuel % Life	Availa. %	Capacity %	Size Mod. Cap., %
1.20	50	923	534	4.88	80.85	70.10	75.04
1.20	33	1382	357	3.26	82.23	71.30	76.40
1.20	25	1850	267	2.43	82.93	71.90	77.10
1.25	100(a)	496	993	9.06	77.30	67.02	64.85
1.25	50	976	505	4.61	81.08	70.30	75.26
1.25	33	1443	342	3.12	82.35	71.40	76.52
1.25	25	1923	256	2.34	83.01	71.97	77.17
1.30	50	986	500	4.57	81.12	70.33	75.30
1.30	33	1471	335	3.06	82.40	71.44	76.57
1.30	25	1955	252	2.30	83.04	72.00	77.21
1.35	50	997	495	4.52	81.16	70.37	75.34
1.35	33	1482	333	3.04	82.42	71.46	76.59
1.35	25	1969	250	2.29	83.06	72.01	77.22
1.40	50	976	505	4.61	81.08	70.30	75.26
1.40	33	1452	340	3.10	82.37	71.41	76.53
1.40	25	1926	256	2.34	83.01	71.97	77.18

(a): Reference Case.

this table require some background discussion about the main factors influencing the plant cycle length and the power produced. First, it is assumed that a standard full-power plant with one-year cycle time (including the refueling outage time) has an availability factor of 75%. The refueling portion of an outage is considered to last 45 days, according to industry surveys(29), although this number is just a "meaningful average", with actual values oscillating around it. The capacity factor of the plant is assumed to be 65%, according to Department of Energy reports(30).

In order to calculate the availability and capacity factors of the reduced-power cores, it is convenient to define some new terms. The first one is the "Active-Time Availability Factor" (referred to from here on as ATAF). This is the availability factor of the plant if the 45 days refueling time are not considered. The conditions affecting the on-line and outage times of the different plants can be considered equal for all plants, except for the incidence of the refueling time, which depends on the length of the core cycle. At this point it is assumed that all plants have the same ATAF, i.e., that all plants have the same amount of non-refueling outages. This assumption should be qualified as very conservative for the low power cores, since they have improved Nuclear Steam Supply System (NSSS) conditions as compared to the standard plant; for the rest of the plant, operating conditions are essentially identical to the standard plant. In actuality, the low power core should have a distinct advantage in savings on unplanned outages.

The ATAF can be calculated from the standard plant assumptions stated above, considering that the product of the ATAF by its corresponding "cycle time" must equal the product of the actual availability factor by the actual cycle time:

$$ATAF * (TCT-ROT) = AF * TCT$$

where ATAF = Active-time availability factor.

TCT = Total cycle time. (365 days for the standard plant).

ROT = Refueling time. (45 days).

AF = Availability factor. (75% for the standard plant).

According to the standard plant assumptions, this definition yields a value for the ATAF of 85.5%.

Another parameter defined to establish the capacity factor variations is termed "Operating Capacity Factor". The ATAF defines the level of on-line time of the plants when the refueling time is not considered. However, even during the on-line time, the plants do not deliver their full power. There are a number of reasons that prevent the plants from delivering 100% power level at all times while they are connected to the grid: One cause is the de-rating of the plant due to technological causes, frequently associated with the fuel and its thermal conditions; another is the power runbacks, or forced temporary de-rating following a power transient, which is normally required in order to assure the fuel and primary coolant system integrity and performance. With these and other similar losses of capacity factor in mind, the average power level of the plant during the on-line time is defined here as

the Operating Capacity Factor. This is another factor that is hereby considered to remain constant for all types of plants for comparison purposes. The operating Capacity factor (which will be referred to as OCF from here on) can be defined and calculated from the assumptions stated above for the standard plant in the following way:

From the definition of ATAF, the on-line time is

$$\text{On-line time} = \text{ATAF} * (\text{TCT} - \text{ROT})$$

where ATAF = Active Time Availability Factor.

      TCT = Total Cycle Time.

      ROT = Refueling Time.

Then, using the concepts of OCF and the conventional Capacity Factor (herein referred to as CF), it is possible to write

$$\text{OCF} * \text{ATAF} * (\text{TCT} - \text{ROT}) = \text{CF} * \text{TCT}$$

From where OCF can be obtained and calculated using the assumptions stated for the standard plant:

$$\text{OCF} = (\text{CF} * \text{TCT}) / (\text{ATAF} * (\text{TCT} - \text{ROT})) = 0.867$$

where OCF = Operating Capacity Factor (Capacity Factor of on-line time).

    CF = Conventional Capacity Factor (Including all time).

One significant factor that can differentiate the plants is the length of the real cycle. Up till now, the burnup calculations

have described the burnup levels achievable and their associated power generation times. However, these are full-power generation times, and it is now necessary to turn them into actual plant times. In a standard plant, the real time or Total Cycle Time and the full-power time are related through the Capacity Factor, so that

$$FPT = CF * TCT$$

where FPT = Full-Power time.

In the case of the low power density cores, it is not possible to use the same concept, since the CF is unknown because of the changing weight of the refueling outage with the different powers and degrees of burnup. However, the OCF defined above can be used to overcome this problem. Using the last two equations,

$$FPT = CF * TCT = OCF * ATAF * (TCT - ROT)$$

Since OCF is considered constant, as well as ATAF and ROT, while FPT is obtained from the burnup studies, it is possible to obtain the total cycle time of any plant as

$$TCT = (FPT / (OCF * ATAF)) + ROT$$

The FPT tabulated in Table 4.2.1 is in hours (from the burnup calculations) while the TCT is more conveniently expressed in days.

Substituting the values obtained for OCF and ATAF allows the above expression to be written as

$$TCT \text{ (days)} = ((FPT \text{ (hours)} / 0.74) + 1080) / 24$$

This total cycle time in days is listed in the third column of Table 4.2.2 for all the cases under study, and is also plotted in Figure 4.2.3 for easier examination and comparison purposes.

Note first that the cycle length shown in Table 4.2.2 for the standard reactor is more than one year. The reason for this is that the burnup levels obtained for the discharged fuel from the calculations are higher than the actual burnup levels of the fuel discharged from real reactors. There are several reasons for this fact. First, the calculational model assumes a slightly high fuel enrichment (later figures seem to point to 3.0% enrichment instead of the 3.1% considered in this study as the equilibrium cycle fuel enrichment). Second, the burnup study assumes uniform burnup of the fuel; the actual fuel burnup distribution is far from being uniform, and this causes the calculational model to sustain criticality somewhat longer than the real core. It should also be noted that the actual average equilibrium cycle time for PWR's in the U.S. during the period 1973 to 1979 was somewhat greater than one year. In any case, even with some possible absolute error associated with the actual values of cycle length, the relative comparison between the different cores should be accurate, since the bias applies to all cores in the same sense.
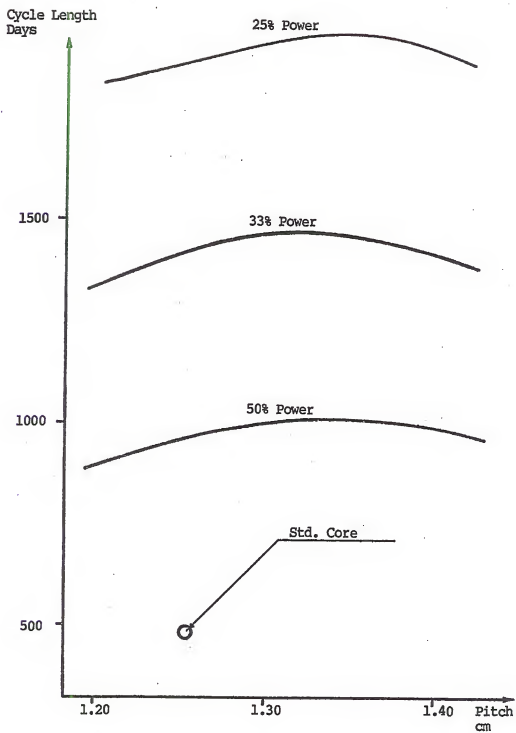
Figure 4.2.3. Real Cycle Length vs. Pitch.

Note that the cycle length of the 50% power, 1.35 cm pitch core is very closely twice that of the standard reactor. This means that if a one-year cycle applies for the standard plant, a two-year term would apply for the 50% power one. This seems obvious, but hidden in here is the fact that for this two year cycle, one refueling outage has been turned into power-producing time. In other words, the refueling outage time is reduced to approximately one-half for this configuration of what it is in a standard plant. This will be seen better in the next column of the table.

The fourth column on Table 4.2.2, labeled Life Reful Days, shows the expected total number of days that the plant would spend in refueling during a thirty year life period. The resulting data are also displayed in Figure 4.2.4 for easier reference and comparison. The purpose of this column is to realize the number of days gained from refueling downtime for power-producing days. The figure is obtained by first finding the number of cycles that the reactor would complete in a thirty year period, and multiplying the number by the forty-five days length assumed for refueling. Note that the 50% power, 1.35 cm pitch reactor shows a savings of 498 days over the life time of the plant, which would represent a considerable amount of money saved.

Another form of looking at the savings in refueling outage time is found in column 5 of Table 4.2.2, labeled Refuel % Life. This column shows the percentage of the total plant life that is spent in refueling. A variation of this figure would be closely
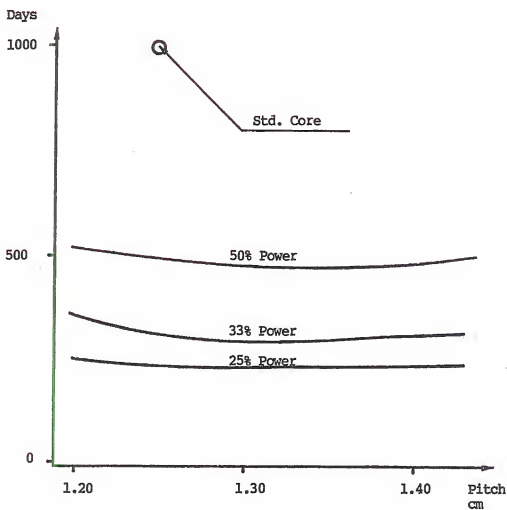
Figure 4.2.4. Total Life Refueling Time vs. Pitch.

related to the variation of the capital costs part of the total cost of energy. It is interesting to note that over 9% of the total plant life in a standard plant is spent refueling, while this figure is reduced to slightly over 4.5% for the optimized 50% power core. and about 3.0% and 2.3% for the 33% and 25% power level cores. Note that the variation of pitches does not heavily affect the percent impact of the refueling time.

Using more conventional parameters, column 6 on Table 4.2.2 labeled Availa. % shows the availability factors that can be expected from the different reactors assuming that they encounter the same percentage of unexpected problems during normal operation (which is a pessimistic assumption when comparing the low power plants to the full power standard plant). The only variable that is supposed to affect the availability of the plant in this calculation is the relative effect of refueling time. The availability factor can be obtained from the formulas defining the Active Time Availability Factor. which was recognized above as an invariable index. The expression for the availability factor is

$$AF = ATAF * (TCT - ROT) / TCT$$

As it was expected, the standard plant shows an availability factor close to 75% (it is slightly higher due to the longer calculated cycle time caused by the reasons stated above). The pitch-optimized, 50% power core shows an availability factor of about 4% over that of the standard plant. while the highest availability. which of course is obtained by a 25% power core. is

just about 2% above that of the 50% power reactor. The availability factors of all the cases considered are plotted as the top set of curves of Figure 4.2.5, plus the point representing the standard core, which is significantly displaced downwards.

Column 7 on Table 4.2.2, labeled Capacity %, is another frequently used parameter for measure of the performance of a plant. In this case, since all plants are so far assumed to have the same amount of unplanned outages, the capacity factor is obtained by use of the Operating Capacity Factor. From the equation defining the OCF, it is possible to write

$$CF = OCF * ATAF * (TCT - ROT) / TCT = AF * OCF$$

As could be expected, the standard plant shows a capacity factor slightly above 65%, while the 50% power, pitch-optimized core holds a capacity factor of about 3.3% above that of the standard reactor, which means an increase of 5% energy production in the same period of time, for equivalent installed powers. The 25% power core capacity factor is increased by only about 1.6% over that of the 50% power core. The capacity factors of all the plants under study are plotted as the bottom set of curves of Figure 4.2.5.

The last column of Table 4.2.2, labeled Size Mod. Cap., %, again calculates the capacity factors that can be expected from the different cores, but this time the calculation takes into account the historically proven fact that smaller size plants actually have better capacity factors than large plants for equally long cycles
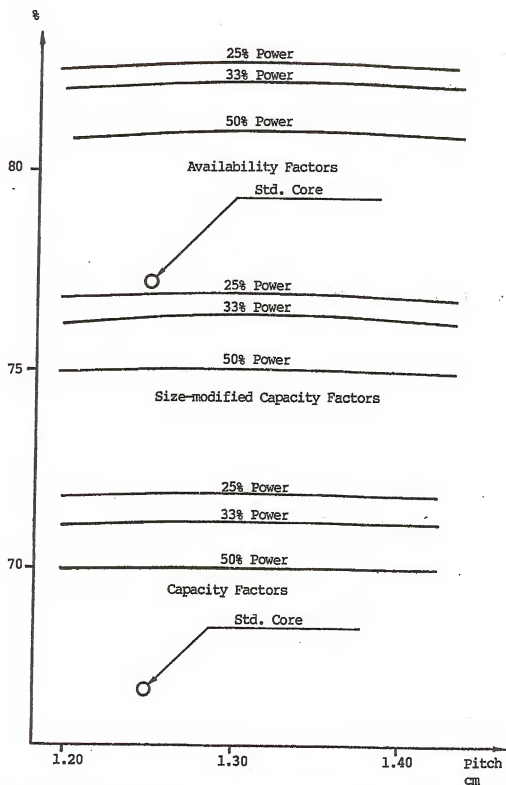
Figure 4.2.5. Availability and Capacity Factors vs. Pitch.

(the standard one-year cycle). The Atomic Industrial Forum(30)
reports capacity factors of 62.3% for large plants (1000 MWe and
above, such as the reference plant of this study) and 69.4 for
plants sized between 400 and 749 MWe. This figure is used here for
the calculation of the capacity factor of all reduced power plants.
Obviously, the capacity factor is again affected by the cycle
length, since this causes different spacings between refueling
outages. The new, "size corrected" capacity factors are then
obtained with the following considerations:

Let TCTo be the total cycle time of the reference plant for
any particular plant size. In this case it is one year, since the
capacity factors collected from real-world data correspond in their
vast majority to yearly cycles. The plant-size dependent Capacity
Factor is called SCF, and is obtained from statistics of existing
different plant sizes.

In TCTo days, there are TCTo * SCF full-power operation days.
During the same time period, the plant is available (TCTo − ROT) *
ATAF days. So, in the TCTo days, there are

$$\frac{TCTo * SCF}{ATAF * (TCTo - ROT)}$$ full-power days per active day.

This is the concept of operating capacity factor, now applied
to any size-dependent actual capacity factor. If the plant can
operate for a known FPT full power time, just as shown before, the
total cycle time should be

$$TCT = (FPT / (OCF * ATAF)) + ROT \quad \text{or}$$

$$TCT = (FPT * (TCTo - ROT) / (TCTo * SCF)) + ROT$$

and then, the capacity factor must be

$$CF = FPT / TCT = FPT / (FPT * (TCTo - ROT) / (TCTo * SCF)) + ROT$$

Since this capacity factor takes into account the variable capacity factor of a certain-size plant for a given standard cycle length, it is called in Table 4.2.2 the Size-modified capacity factor. The size-modified capacity factor is also plotted in Figure 4.2.5, as the middle set of curves in the figure.

Note that when this size correction is taken into account, the 50% power, 1.35 cm pitch reactor shows a capacity factor over 10% above that of a standard plant which represents a 16% increase in the power being produced in equal amounts of time by equivalent power installations. This is a very significant increment, and its economical impact is bound to be of large importance because it reduces the impact of the capital and O&M costs on the total cost of power, and in the case of nuclear-generated power, these two items are responsible for a large proportion of the energy cost. Under the present assumptions, the 50% pitch-optimized power plant would reach capacity factors of about 75%. Also, in net energy generation, an optimized SHARP will be closer to the output of a standard size plant than its rated power indicates.

4.2.3. Ore Usage and Enrichment Needs.

The last comparison used in the study is complementing the information shown in Section 4.2.1, but establishing a common baseline on which to compare all the different plants.

Table 4.2.3 refers to the usage of ore and enrichment units for the different plants considered, but with all data normalized to the power generation of 1125 MWe, so that the figures on the table can be compared on a same energy output basis. As in Tables 4.2.1 and 4.2.2, the two first columns are used for core identification. The third column shows the total mass of plutonium discharged yearly. It is assumed here that the total energy production of 1125 MWe over the 30 year life time is 22.4 GWy. Then, the plutonium mass discharged yearly by each core, for an energy production equivalent to one year of operation of the standard plant is obtained as

$$\text{Disch. Pu (Kg)} = (\text{MTPu/MWd}) * 22400*30*365.25 / 0.33$$

where MTPu/MWd is the total plutonium mass discharged per MWd thermal produced. This figure was calculated for each core in Table 4.2.1.

As could be expected, the plutonium discharge is reduced as the fuel pitch becomes larger. The cause is the softening of the neutron spectrum due to the better moderated geometry obtained with the increased amount of water present in the fuel cell. For each particular pitch, the plutonium yield is also reduced as the power level decreases, although the effect is much milder than the one

Table 4.2.3. Ore, Enrichment and Plutonium Normalized to 1125 MWe

Pitch cm	Power %	Pu Disch. Kg/Yr	Ore Usage MT/Life	Ore Savings % Over Std.	Enrich. Need SWU/Life
1.20	50	225.08	4662	−2.87	3158
1.20	33	219.69	4589	−1.28	3109
1.20	25	216.87	4533	−0.05	3071
1.25	100(a)	201.72	4531	0.00	3070
1.25	50	197.51	4396	2.98	2978
1.25	33	194.83	4389	3.15	2973
1.25	25	192.61	4357	3.86	2952
1.30	50	181.48	4350	4.00	2947
1.30	33	178.07	4304	5.03	2916
1.30	25	176.21	4284	5.46	2902
1.35	50	169.45	4300	5.12	2913
1.35	33	166.30	4270	5.76	2893
1.35	25	164.66	4254	6.12	2882
1.40	50	160.95	4395	3.01	2978
1.40	33	158.33	4361	3.75	2955
1.40	25	156.88	4349	4.02	2947

(a): Reference Case.

associated with the change of pitch. The cause of the spectral shift associated with the power level variation is the change of the Doppler broadening of resonances, which is decreased as the fuel temperature decreases. These two effects add up to reduce the plutonium discharge by 16% for the 50%, 1.35 cm pitch core, as compared to the standard reactor. This is obviously a favorable point for the low-power core from the non-proliferation point of view.

The fourth column on Table 4.2.3, labeled Ore Usage MT/Life, shows the total amount of U3O8 ore used by each core for the production of 22.4 GWy (e) of energy. This is, as mentioned above, the total energy production assumed in the lifetime of the standard plant. An overall plant efficiency of 33% was assumed for all plants, since the thermodynamic conditions are supposed to be identical. Since Table 7.2.1 showed the U3O8 needs per MWd (th) produced, those figures can be used in order to obtain the lifetime ore consumption as

$$\text{Ore consumption (MT)} = 2.45448 \text{ E+7} * \text{(MT/MWd th)}$$

where MT/MWd th is the specific ore consumption shown on Table 4.2.1. Note that the specific ore consumption was calculated under the assumptions of 3.1% enrichment in U-235 of the fresh fuel and 0.2% enrichment of the enrichment plant tails. Obviously, this column shows the ore usage under the assumption of once-through fuel management.

Note that savings of 5.1% of ore can be achieved by the 50% power core with the optimized pitch. The ore usage figures follow

a parallel trajectory to the discharge burnup figures. For each pitch, reducing the power level reduces the ore requirements, but as pitches are changed, a minimum ore requirement is reached for 1.35 cm pitch.

The comparison of ore usage with respect to the standard plant is made in column 5 of Table 4.2.3. under the label Ore Savings % Over Std. It is easier to see in this column that large savings are achieved with the reduction of power from 100% power level to 50%, but further power reductions result in additional ore savings of a much smaller magnitude. per unit of power reduction. This is one of the main reasons calling for the power reduction to be carried to no more than 50% of the standard value. Actually, an optimum SHARP power level will be higher than 50%. and will be affected by other economic considerations.

Column 6 of Table 4.2.3 shows the total enrichment needs of each plant. in SWU. This column assumes also the total energy production of all configurations to be 22.4 GWy (e), the fresh fuel enrichment to be 3.1%, and the tails assay to be 0.2%.

5.1. Safety Related Thermal-hydraulic Considerations

5.1.1. Motivation

As indicated in the previous chapters, the reduced-power cores that would be installed in a SHARP would have very nearly the same physical design as the standard full-power reactor, except for some possible variation in the fuel array pitch. It has also been stated that the balance of the plant must operate at conditions essentially equivalent to those of the standard reactor, in order to maintain thermodynamic efficiency, and to avoid the need for significant redesign of components in the balance of the plant.

The above conditions require the reactor coolant to have identical thermodynamic characteristics at the inlet and outlet of the reduced-power core as it would have with the standard reactor. It is obvious then, that if the core is operating at a reduced power level, the coolant flow must be reduced accordingly in order to obtain the same enthalpy at the outlet. There is, however, a consideration of major importance related to reducing core flow rate: the heat transfer conditions are affected by the different power density and by the different coolant flow conditions. This presents a most important issue from the standpoint of plant safety, since it is well known that one of the critical

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technological and safety aspects of the standard plant is the assurance of an adequate heat transfer under normal, transient and accident conditions. Mark Miller performed a detailed study(16) on the behavior of the low-power core as compared to the standard reactor and some of the main conclusions of this study are presented here as illustrations of the thermal-hydraulic and safety comparison between the two reactor concepts.

5.1.2. Thermal-hydraulic Studies.

There are several parameters that are considered of key importance for thermal-hydraulic safety considerations. Some of these are the fuel average temperature, which is an indication of the stored heat content of the core and which is an important parameter in determining the emergency core cooling system characteristics; the fuel centerline temperature, of particular importance since that is the hottest point in the pellet, the clad surface temperature, and the departure from nucleate boiling ratio (DNBR) which is the ratio between the surface critical heat flux (the surface heat flux that would produce film boiling) and the actual surface heat flux.

The DNBR varies from point to point in the reactor, since the heat transfer conditions change from one point on the fuel to another, as does the heat flux level. A safety margin can be related to the minimum DNBR found in the whole core. Thus, the minimum DNBR is one of the criteria that can be used for thermal-hydraulic safety comparisons. The importance of the clad

surface temperature as an indication of core safety becomes apparent when considering the possibility of clad failure. The clad surface temperature depends on the heat flux level and on the thermodynamic conditions of the coolant. Although the coolant is kept by definition at equivalent conditions in all the cores under study, the heat flux experiences significant differences, which may result in changes in the clad surface temperature. Finally, the fuel temperature depends not only on the thermodynamic coolant conditions and on the clad-to-coolant heat transfer mode, but it depends to a high degree on the heat generation rate at each point of the fuel pins. The heat generation rate is different from point to point in the core, and it is related to space-dependent nuclear properties of the fuel and the core itself: the particular fuel composition at each point (which depends on the power generation history of that point), the poison concentration, the proximity to water holes or control rods, etc. all contribute to an uneven distribution of the heat generation rate throughout the core. This unevenness has to be taken into account when studying the limiting heat transfer and temperature conditions, and it is done by means of the so-called power peaking factors. It is normally assumed that the heat generation rate has an axial variation similar to a cosine function; a radial variation due to different geometrical factors is also assumed. The power peaking factors used for the present calculation are 1.64 for the radial value, and 2.62 for the overall core. These power peaking factors and the thermal-hydraulic safety parameters stated above are included in

the TEMPRET(31) code, developed by Miller and based on a combination of industry-accepted heat transfer correlations, which was used for the safety-related thermal-hydraulic calculations.

Figure 5.1.1 shows the fuel centerline temperature at the most disadvantageous point, as a function of the core power density level and of the coolant mass flow rate. The relative mass flow rate (on the x-axis) is referred to the standard core's nominal mass flow rate. The marked points correspond to the flow rates that yield outlet coolant enthalpies equivalent to those found in the standard core. Note that for high mass flow rates (as compared to the value required for equivalent outlet enthalpy), the heat transfer takes place as forced convection, and is dependent on the flow conditions. Once a certain low mass flow rate is achieved, nucleate boiling takes place, and the heat transfer is essentially independent of the flow conditions. The fuel centerline temperature remains at an essentially constant level as the coolant flow is further reduced. This situation would hold as long as the coolant mass flow rate is not reduced below the point where film boiling would start. Figure 5.1.2 is a plot similar to Figure 5.1.1 but it represents the volume averaged temperature of the fuel. The same trends can be observed as in Figure 5.1.1, but they are smoothed out by the averaging process, which for each flow rate includes portions of fuel that are under forced convection conditions as well as other portions which are under nucleate boiling conditions.
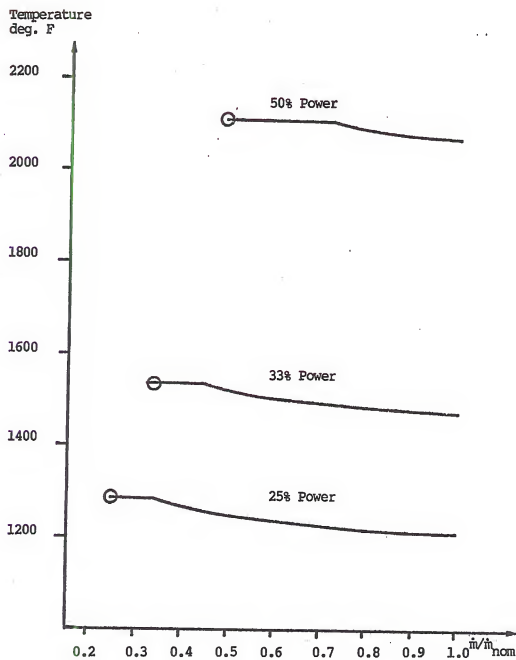
Figure 5.1.1. Fuel Centerline Temperature vs. Relative Mass Flow Rate

Figure 5.1.3 shows the fuel clad surface temperature as a function of the the coolant flow rate for different fractions of the standard power density, for both the maximum clad surface temperature point, and for the point corresponding to the maximum heat flux rate. The transition from the forced convection to the nucleate boiling conditions is more apparent here, since the clad surface temperature is the parameter most affected by this heat transfer condition. The sharp break point displayed in the curves results from changes in the type of correlation used for computing the temperature level. Note that at the normal operating flow rate for each particular core, all cores have practically the same maximum clad temperature, regardless of the fraction of standard power level considered.

Figure 5.1.4 shows the minimum DNBR for the three low power density cores as a function of the coolant mass flow rate. Again, the marked point on each curve represents the mass flow rates that correspond to the standard outlet coolant enthalpy. The graph also shows the minimum DNBR level of the standard full-power reactor. Note that all the low power density cores have an advantage with regard to the minimum DNBR when compared to the standard core, for equivalent coolant conditions. However, it is important to keep in mind that Figure 5.1.4 corresponds to cores with the standard fuel lattice pitch. One should recall that the burnup studies showed that a pitch of 1.35 cm is advantageous from the fuel utilization point of view as compared to the standard 1.2573 cm pitch. It is important to examine the thermal-hydraulic behavior of the cores at
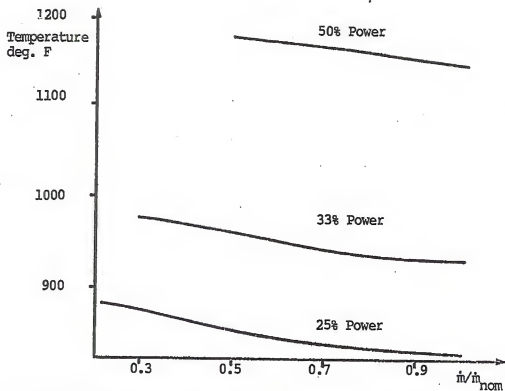
Figure 5.1.2. Fuel Average Temperature vs. Relative Mass Flow Rate.
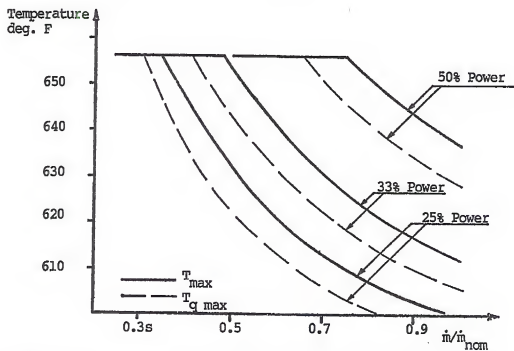

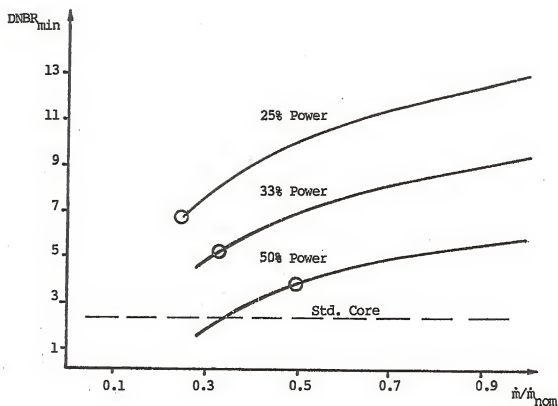Figure 5.1.3. Clad Surface Temperature vs. Relative Mass Flow Rate.

Figure 5.1.4. DNBR-minimum vs. Relative Mass Flow Rate.

that burnup-optimized pitch configuration, as well as at the standard value. Figure 5.1.5 shows the results of these calculations.

Figure 5.1.5 is a plot of the percent savings relative to standard core values achieved in the minimum DNBR, and in the maximum and average fuel temperatures as a function of the core power density level. Both the standard fuel pitch and the burnup-optimized fuel pitch have been considered. These calculations assume a coolant flow rate that maintains the outlet coolant enthalpy at a the nominal standard core level.

When the standard pitch is considered, reducing the power density causes a constant improvement in the minimum DNBR, tending to an infinite value as the "zero-power" reactor is approached. Obviously, the savings for the 100% power level is zero. Both the maximum and the average fuel temperatures show an improvement, but it is clearly bound, and tends to level off. Note that the fuel temperatures (maximum or average) are not affected by the pitch value.

When the burnup-optimized pitch is considered, the same coolant mass flow requires a lower fluid velocity, which tends to worsen the heat transfer conditions. This fact is apparent in the corresponding DNBR curve, which appears shifted downwards from the one corresponding to the standard pitch. This is the fact that prevents the standard full-power reactor from operating at the burnup-optimized fuel pitch, as was indicated in the preliminary calculations chapter. The minimum DNBR would be 20% worse than it
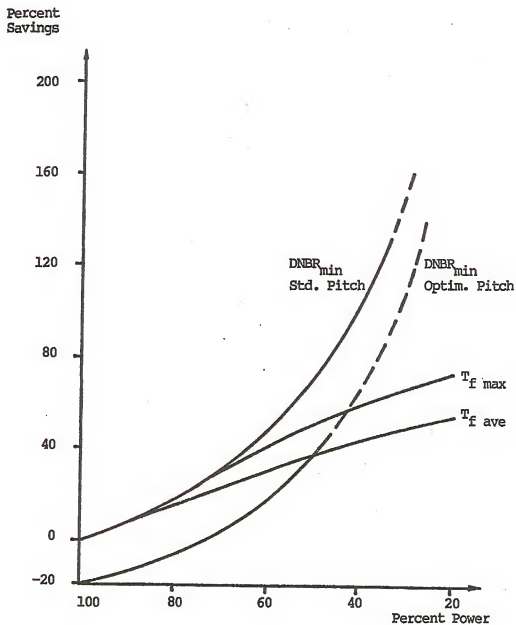
Figure 5.1.5. Percent Savings in DNBR-minimum and Fuel Temperature.

is in the standard reactor. which is unacceptable from a fuel safety standpoint. Further examination of the curve shows that at a power density level of about 75%. the minimum DNBR becomes equivalent to that of the standard reactor with the standard pitch. Any power density level chosen below 75% can be used with the burnup-optimized fuel pitch value. and it still presents improved heat transfer safety conditions as compared to the current standard reactor.

It is important to remember that the thermal-hydraulic safety conditions examined here correspond to the steady-state operation of the core. The transient analysis being conducted for these cores are still under preparation at the time of this report. Some of the transient conditions that should be examined are an overpower transient. a partial and a total loss of coolant flow with or without power reduction. and a depressurization transient. It can be stated that the low power density cores show strong advantages over the standard core with respect to thermal transient behavior. The reduced fuel temperature allows for a larger amount of heat to be stored in the fuel before critical heat transfer conditions are reached. This fact. combined with the lower power density level of the cores. allows for significantly longer reaction times before any damage occurs to the fuel. In transient situations or in loss-of-coolant flow accidents. the larger burnup-improved pitch would seem to have a slight advantage again. since it provides a higher inventory of water and larger coolant channels in the core than the standard pitch. This will tend to

reduce the probabilities of fuel dryout and the consequential cladding failure.

5.2. Economic Evaluation

5.2.1. Introduction.

The technical aspects surveyed so far about the SHARP clearly show its advantages over the standard plant in the aspects of operational safety, once-through fuel utilization, high radioactive waste storage and disposal, reduction of refueling outages and associated personnel radiation exposure, etc. However, it is evident that the SHARP can be viewed as an "oversized" plant, requiring more fuel inventory than standard power density cores for an equivalent amount of output power, as well as having some of the plant equipment larger than would be necessary at standard power densities for the production of a given amount of power. All these considerations indicate that the capital cost of the SHARP will be larger than that of the standard plant. There is, however, the belief that the reduction of refueling time and improved fuel utilization will help in reducing the impact of the increased capital cost. It thus becomes obvious that an economical comparative analysis is needed in order to asess the final cost of energy produced by the SHARP versus the cost of power produced by a standard plant.

The cost of energy produced by a nuclear power plant, is typically broken down into three main components: capital cost, fuel cost, and operations and maintenance cost. In the case of the SHARP, it might be possible to think that the enhanced safety could

result in some monetary savings due to reductions in licensing times, in plant construction, etc. but these are rather speculative, and they are considered out of the scope of this report. The economical comparison presented here is based on the costs of fuel, capital and operations and maintenance of several different power density plants. A more detailed definition of the sizes of the main plant components for each reduced-power density plant is given in next section.

5.2.2. The Economic Comparison Studies.

The economic comparison studies are presented for a set of five different plant output powers. They correspond to 35%, 50%, 60%, 70%, and 100% of the standard plant power as defined in Chapter II. Each of the plants considered in this study is composed of a full size Nuclear Steam Supply System (NSSS) (as used for the standard plant); a Balance Of Plant (BOP) of the nominal size corresponding to the plant output power; and finally the containment building sized for 120% of the size that would correspond to a standard power density plant of the same power rating. All plants are assigned a 30 years life, and startup is assumed in 1993. Costs are levelized to and expressed in 1993 dollars.

Three possible inflation scenarios are considered for each case studied. The low inflation scenario assumes an inflation rate of 5% per year; it assigns an interest rate of 7% for borrowed funds and bonds, and a 13% rate of return is assumed for common equity (stocks). The moderate inflation scenario assumes a yearly

inflation rate of 9%; the assigned interest rate for borrowed funds and bonds is 11%, and the rate of return on common equity is 17%. Finally, the high inflation scenario assumes a yearly inflation rate of 13%; it assigns an interest rate of 15% to borrowed funds and bonds, and the return on common equity is placed at 21%.

The economic calculations were performed by Hersperger(1) with the help of three industry accepted codes: CONCEPT-IV(32) for the calculation of the capital cost associated with each plant; GEM(33) for the calculation of the fuel cycle costs; and POWERCO(34) which computes the total generation costs. A brief description of the three codes is found in Appendices B9, B10 and B11, respectively.

Table 5.2.1 presents the capital costs of the different plant sizes and the percentage increase with respect to the cost of the standard plant, for the three inflation scenarios defined. It is evident that the capital cost per KWh is a uniformly increasing function, whose slope becomes increasingly steep as the plant power level is reduced. Any power reduction beyond about 50% of the standard becomes absolutely impractical since the capital cost is a very significant proportion of the total power generation cost. The trends of capital cost versus plant power rating for the different inflation scenarios are plotted in Figure 5.2.1.

Table 5.2.2 presents the fuel costs for the different plant sizes and inflation scenarios. The percentage increases of cost with respect to the standard plant fuel cost (for each particular inflation scenario) are also shown. These fuel costs include all fuel-related expenses, such as mining, enrichment, fabrication,

Table 5.2.1
Capital Costs (Mills/KWh) and Percent Increases over Standard Plant

Power Level	Low-inflation		Moderate-inflation		High-inflation	
	Cost	% Inc.	Cost	% Inc.	Cost	% Inc.
35%	50.22	73.3	108.64	72.6	221.07	72.0
50%	40.89	41.1	88.59	40.8	180.55	40.5
60%	37.10	28.1	80.43	27.8	163.98	27.6
70%	34.37	18.6	74.62	18.6	152.33	18.5
100%	28.97	0.0	62.94	0.0	128.52	0.0

Table 5.2.2
Fuel Costs (Mills/KWh) and Percent Increases over Standard Plant

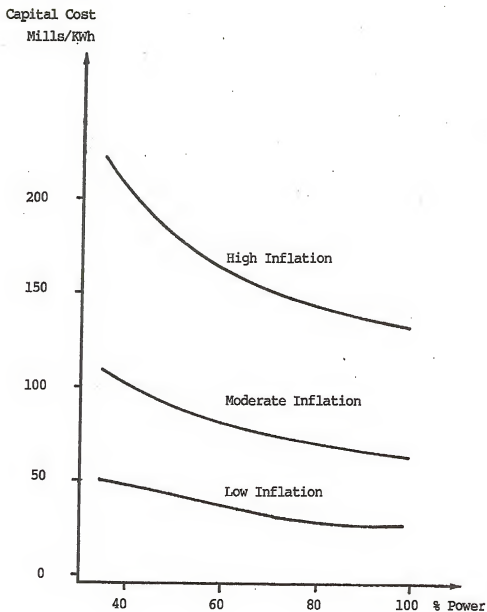| Power Level	Low-inflation		Moderate-inflation		High-inflation	
	Cost	% Inc.	Cost	% Inc.	Cost	% Inc.
35%	26.52	22.4	54.95	15.9	100.32	4.7
50%	24.87	14.8	53.30	12.4	103.70	8.2
60%	23.73	9.5	51.24	8.1	101.14	5.5
70%	22.93	5.8	49.78	5.0	99.18	3.5
100%	21.67	0.0	47.41	0.0	95.85	0.0

Figure 5.2.1. Capital Cost vs. Plant Power.

Note: Power is expressed as percent of Standard Plant.

etc. up to transportation and storage of spent fuel. As with the rest of this economic evaluation. all plants are supposed to start operation in January 1993. and the costs are expressed in 1993 dollars. Note that the fuel cost is one item that was expected to improve the economics of the low power density plants due to increases in burnup. However. they all show a larger fuel cost when compared to the standard plant. because it takes much longer for a given batch of fuel to produce its share of energy. This means that in a standard plant. a batch of fuel has produced all its energy (and therefore has produced all its income) in a three-year period; in the case of a 50% power plant. it takes more than six years for the same batch of fuel to produce all of its energy. The income is therefore retarded, and the capital invested in the fuel batch suffers higher interest charges, which outweigh the relatively small increment of burnup that is obtained from the fuel.

It is clear from the figures on Table 5-2.2 that the turnout time for the fuel costs is important for the economy of the fuel cycle. This strong sensitivity to time suggests that it might be an advantage for the SHARP to consider low enrichment fuels, which would have reduced enrichment costs and would result in shorter core cycles. On the other hand, studies performed in the industry(3) show that reducing the enrichment results in a worse ore utilization. These two considerations indicate the need for a more complete study of the economic behavior of the SHARP's as fuel enrichment is allowed to vary. Figure 5-2.2 is a plot of the : ..
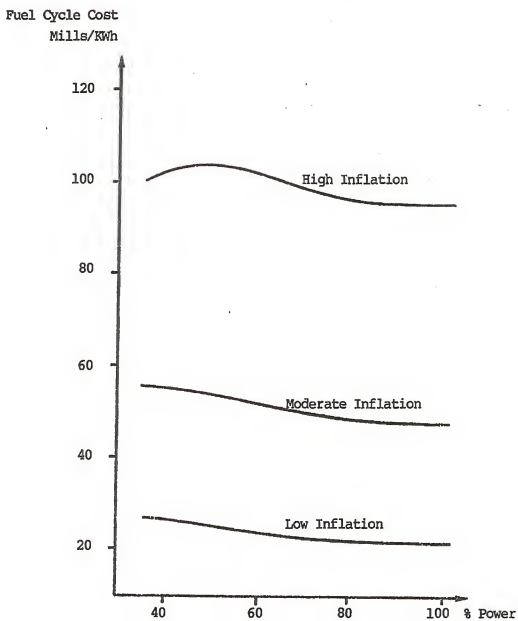
Figure 5.2.2. Fuel Cycle Cost vs. Plant Power.

Note: Power is expressed as percent of Standard Plant.

fuel cycle cost vs. plant size, for the three inflation scenarios being considered.

The operations and maintenance (O&M) costs are considered constant for all the plants studied. They are fixed as 7.18 mills/KWh for the low-inflation scenario, 11.25 mills/KWh for the moderate-inflation scenario, and 17.34 mills/KWh for the high-inflation scenario. These values have been calculated from present average values for nuclear plants, and modified with an inflationary factor. Note that the O&M costs should be dependent on plant size, since the burnup studies showed drastically different core cycle lives depending on the power density level. However, the O&M part of generation cost is a very small proportion of the total power generation cost, and the speculative differences that could be applied would not affect the final generation cost but by some tenths of a percent. It should be realized, however, that the longer cycles and reduced refueling outages have been taken into account in the evaluation of the fuel cycle costs. One element that has not been considered is the cost of replacement power for the different refueling downtimes. Although their cost is presently very high due to the fact that most replacement power is produced by oil or gas units, it is difficult to forecast their cost at significantly later times. In any case, this could account for an additional one or two percent reduction of power generation cost for the low power density plants, which has not been accounted for in the figures presented here.

Table 5.2.3 contains the total generation costs of the plants under consideration, for the three inflation scenarios, as well as their cost increment with respect to the full-power plant. Note that like the capital cost, and unlike the fuel cost, the total generation costs exhibit percentage increases at reduced power which are essentially independent of the inflation level. This table is the most significant one from the economic viewpoint, since it shows the cost of the energy from each plant configuration, after all parameters have been taken into account. Power levels between 60% and 70% of the standard level show moderate total cost increases. The transition from 60% to 50% starts to show a sharper increase in generation cost, and it is evident that power levels below 50% can not be considered since their economic penalty is unreasonably large. These trends can be better observed in the curves shown in Figure 5.2.3, which are a plot of the total generation cost vs. the plant size, for the three inflation scenarios.

Table 5.2.3
Generation Costs (Mills/KWh) and Percent Increases over Standard Plant

Power Level	Low-inflation		Moderate-inflation		High-inflation	
	Cost	% Inc.	Cost	% Inc.	Cost	% Inc.
35%	83.92	45.1	174.84	43.8	338.73	40.1
50%	72.94	26.1	153.14	25.9	301.59	24.8
60%	68.01	17.6	142.92	17.5	282.46	16.9
70%	64.48	11.5	135.65	11.5	268.85	11.2
100%	57.82	0.0	121.60	0.0	241.71	0.0
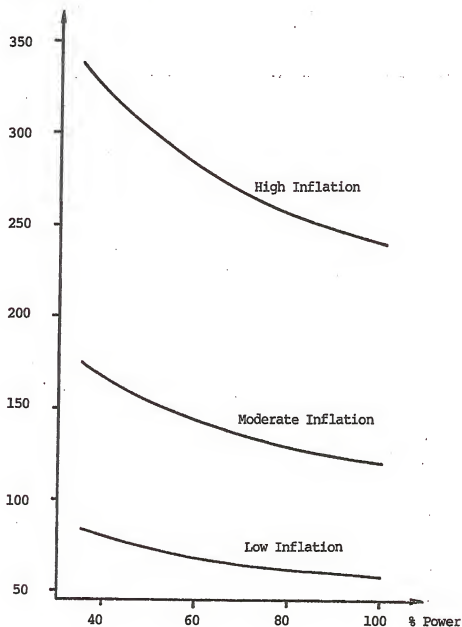
Figure 5.2.3. Total Generation Cost vs. Plant Power.
Note: Power is expressed as percent of Standard Plant.

CHAPTER VI
CONCLUSIONS AND RECOMMENDATIONS FOR FURTHER RESEARCH

6.1. Introduction

The primary purpose of this work is to establish the neutronic and fuel utilization characteristics of the SHARP (Safer – Highly Available Reactor Plant), as well as to develop an accurate and cost-effective calculational tool for the scoping studies of fuel burnup. In addition, the primary thermal-hydraulics, safety and economic characteristics of the concept and its advantages and disadvantages over the present standard PWR reactor plant are analyzed and summarized to provide a comprehensive review of the key issues needed to judge its technical and economic merits.

The SHARP concept uses a standard, large size PWR reactor with minimal design alterations to drive a reduced power rating electric generation plant. The reactor is used at a reduced linear power, while maintaining coolant thermodynamic characteristics in order to keep the steam cycle efficiency. The immediate result of the power density reduction is a significant improvement in technological operational limits and an overall enhancement of reactor safety, although it also results in a capital cost increase per unit of installed power. The reduced power density entails differences in the nuclear characteristics of the core, which require a thorough study and optimization.

Standard industry neutronic codes (like PHROG, BRT, MONA and LEOPARD) were used for the scoping studies conducted in order to pinpoint the key differential characteristics of the SHARP core as compared to the standard PWR core, and to determine the type and range of parameters on which to conduct the in-depth optimization study. One of the key issues distinguishing the SHARP concept is the improvement of burnup and ore utilization, and this aspect required thorough analysis. The available burnup calculational methods were reviewed and found to be inappropiate for the type of study required for the in-depth optimization stage of the SHARP investigations. The problems with existing burnup calculational schemes was that either computational requirements and expenses were too large or accuracy was insufficient.

A new computer code called CRIBUR was developed for studying the burnup performance of PWR's with a reduced computational cost and a significantly increased accuracy as compared to previous schemes of similar cost. CRIBUR is used in conjunction with accepted industry codes (EPRI-CELL, NUPUNCHER, PDQ-7) to complete the burnup calculational scheme. The final set of codes allows the evaluation of isotopic composition through the fuel life, is able to assign time-varying power levels to the different fuel batches present in the core, and performs the burning of the fuel in a core environment that closely simulates the criticality condition of the reactor through life. The last feature increases the accuracy of this type of burnup calculational scheme because it improves the calculation of the absorption cross section of the moderator

medium; it therefore leads to an improved determination of the neutron spectrum and the isotopic dynamics. The time-varying power level capability is an unusual feature for this type of burnup calculational tool. and has a significant impact on spectrum-to-burnup level correspondence.

CRIBUR results were benchmarked against detailed two-dimensional neutron diffusion theory codes and against actual core data and were shown to consistently yield better results than other popular calculational methods like the ones using batch-averaging for neutron multiplication factor calculations(28), or the ones using the linear reactivity assumptions for end-of-life estimates(15). CRIBUR criticality calculations agree with actual core EOL data to within 1.3%. Batchwise power distribution calculations agree with multi-dimensional core follow calculations to within a few percent.

The computational cost of the burnup calculational scheme is little affected by CRIBUR. since it only takes a minimal fraction of the computer time needed for EPRI-CELL and PDQ-7 calculations which are required to prepare the batch burnup data for CRIBUR. One complete core burnup calculation can be performed with one EPRI-CELL pin life calculation (requiring about three minutes of CPU time of the IBM 370-3033 unit). one NUPUNCHER calculation (requiring about ten seconds). three 1/4-assembly life burnup PDQ-7 calculations (requiring approximately one minute of CPU time each on the same unit) and three CRIBUR calculations (each requiring just a few seconds of computer time).

6.2. Neutronics, Burnup and Ore Usage

The in-depth burnup studies were carried out on a series of cores in which power density and fuel lattice pitch were used as variable parameters. The standard Westinghouse 4-loop 17X17 fuel assembly PWR core was used as a reference core against which the SHARP results were compared. The reduction of power density causes a decrease of Doppler resonance absorptions in the fuel and a reduction of xenon concentration level, resulting in additional reactivity being available for further burning of the fuel. The variation of fuel lattice pitch was shown to have an even larger effect on discharge burnup than the reduction of power density level. These trends can be observed in Figure 6.2.1. It appears that the initial power level reduction (from 100% to 50% of the standard core power level) results in a higher increase of burnup per unit power reduction than further power level reductions to 33 or 25% of the standard reactor's power. The burnup increments per percent power reduction are 0.11% for the 100% to 50% power reduction, 0.092% for the 50% to 33% reduction and 0.087% for the 33% to 25% power reduction. These figures clearly indicate that the effect of power density level reductions on burnup are more beneficial for the initial reductions. This factor, favoring smaller reductions in power density, is further reinforced by the fast deterioration of the economic aspects of the plants with power density levels below 50% of the standard plant.

Examination of the effects of fuel pitch variations shows that all power density level cores have a maximum discharge burnup level
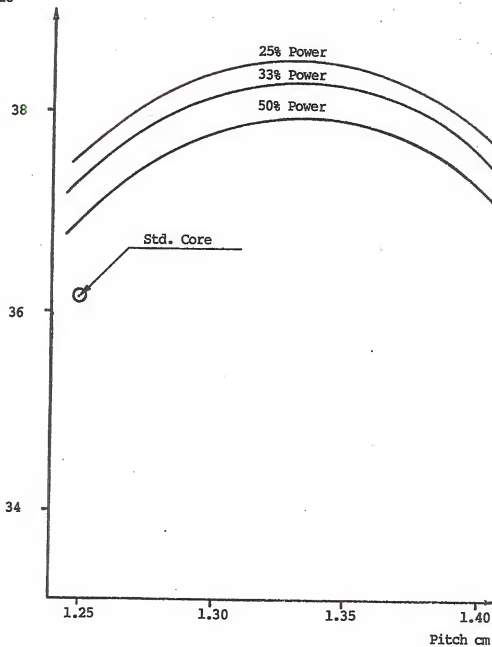
Figure 6.2.1. Discharge Burnup vs. Pitch.

Note: Data from EPRI-CELL, NUPUNCHER, PDQ-7, CRIBUR Burnup Calculations.
Power Levels Indicated as Percent of Standard Core Power.

at a pitch of 1.35 cm. This burnup-optimized pitch can not be used by the standard reactor for thermal-hydraulic safety reasons, but presents no problem for the SHARP's using a power density level of 70% of the standard value, or less. The additional burnups achieved by the SHARP's fall well within the range of acceptable burnup level for the present fuel technology, and thus present no need for clad structure or pin geometry redesign.

The burnup-optimized pitch is smaller than the optimum moderated pitch (which is about 1.6 cm). This is due to the sharp reduction of plutonium conversion associated with the softer spectrum present in the better moderated pitches. Plutonium discharge per unit energy produced from the burnup-optimized 50% power density SHARP is 16% lower than from the standard PWR. This is a positive effect from the nuclear weapons non-proliferation viewpoint. However, in a fuel reprocessing policy environment, such a low plutonium discharge would result in a loss of spent fuel value.

Under the present once-through fuel policy, the SHARP improves ore utilization as a result of the improved discharge burnup capabilities. The 50% power core with the optimized pitch shows a 5.4% better ore utilization than the standard PWR plant. The 33% power and 25% power SHARP's show ore utilizations which are 6.1% and 6.5% better than the standard plant, respectively.

With spent fuel reprocessing in mind, the net consumption of fissile material per unit power produced was calculated. This took into account the fissile mass loaded with the fresh fuel as U-235

and the discharged fissile mass, composed of U-235 and plutonium. Fuel lattice pitch appears as a key factor influencing this variable; the better moderated pitch configurations show a sharp increase in ore usage per unit energy generated due to the reduction of plutonium conversion. The 50% power, burnup optimized pitch SHARP shows a 10% increase of net fissile species usage with respect to the standard plant. However, the power density level shows practically no influence in net fissile species consumption; the 50% power level SHARP using the standard reactor pitch shows a difference of less than 1% with respect to the standard plant.

6.3. Plant Operations Considerations

The reduction of linear power generation entails significant changes in the plant's operational characteristics. The two key effects are the improved safety of the reactor and the increase of the core cycle life. The heat transfer conditions at the pin surface are drastically ameliorated by the reduction in the linear power. Core minimum DNBR savings of 60% with respect to the standard core value are obtained in a 50% power density core using the standard fuel pitch. The same core presents approximately 30% minimum DNBR savings when using the burnup-optimized pitch. Using the burnup optimized pitch in a standard power density reactor would result in an unacceptable 20% worse minimum DNBR. The power density reduction has been also shown to reduce the fuel average temperature by 300 deg. K, resulting in improved fuel conditions, including a reduction of gaseous fission product gas migration.

Another important improved operational variable is the xenon concentration level. A 50% power level SHARP has an equilibrium xenon concentration reduced by more than 25% with respect to the standard plant value. This implies significant reduction of problems associated with xenon oscillations or xenon concentration transients caused by reactor shutdowns or large power variations.

The SHARP concept results in largely increased core cycle lives as a result of two adding factors: the reduction of power level itself, which entails a longer time in order to produce a given amount of energy from the fuel, and the associated discharge burnup level increase, which further contributes to the core life. One implication of these longer core cycles is the reduction of refueling outage time, and its associated high power replacement costs. The 50% power, pitch optimized reactor shows a savings of about 50% in total number of refueling outage days in the plant life. A consequential effect is the increase of the plant availability and capacity factors. The 50% power, pitch optimized reactor shows a plant capacity factor increase of about 3.5% with respect to the standard plant under the assumption of equal probability of unplanned outages. When historical data concerning reduced size plants was reviewed, it became apparent that these plants have actually less unplanned outages than the large ones. Evaluations of plant capacity factors based on forced outage statistical data indicate a capacity factor increase of slightly over 10% with respect to the standard plant for the 50% power, burnup optimized SHARP.

Another consideration of importance from the operations point of view is the likelihood of reduction of annual allowed radiation exposure doses for plant personnel, which would have a large effect on the operation of nuclear plants. The exposure level can be significantly reduced by the SHARP due to the reduction of refueling and forced outages.

The SHARP concept appears clearly as having significant advantages over the standard plant from the operations point of view.

6.4. Economic Effects

The SHARP concept presents both positive and negative factors from the economic viewpoint. The reduction of refueling outage time is an asset because of better utilization of the large capital investments resulting in a reduction of the net energy generation cost. The replacement power cost was not taken into account in the present economic study because of the uncertainties in determining it in a long term extrapolation, which is heavily influenced by the particular characteristics of each utility. In any case, this is a factor that could further improve the SHARP energy production cost. The increase of plant capacity factor was taken into account in the economic performance studies.

The increase of discharge burnup level results in direct although small savings in ore and fuel enrichment and handling costs.

On the negative side, the SHARP present a much higher capital cost per unit installed power, which ranges between 23% and 41% above that of the standard plant, for power density levels between 65% and 50% of the standard value. Inflation level variations have little influence on the relative capital costs when comparing SHARP's to standard power density level plants of equal output rating. The fuel cycle cost analysis shows that in spite of the reduced ore and enrichment needs, the total fuel cost is higher for the SHARP than for the standard plant. The main reason for this behavior is the higher interest costs associated with the long core residence of the fuel. The fuel costs for the SHARP range between 5% and 15% above those of the standard plant's for the power density range defined above. Inflation level has a significant effect on fuel cycle cost, reducing the difference between the SHARP and the standard plant as inflation rates become higher. This is due to differences in revenue schedules on the cost of fuel shipment and storage.

The overall economic evaluation of the SHARP indicates generation cost increases for the SHARP between 17% and 27% with respect to the standard plant for power levels between 65% and 50% of the standard plant. Considering these results, a power reduction beyond 50% of the standard value appears to be an undesirable arrangement when cost/benefits are considered. The combined consideration of the economic and safety aspects shows that a power density level between 50% and 65% of the standard plant would result in improved safety with respect to the standard

plant, and with a generation cost increase of about 20%, with a slight tendency to decrease for higher inflation rate scenarios.

6.5. Suggestions for Future Research

The calculations performed in this study showed the significant potential for operational and safety improvements in PWR plant performance under the SHARP concept. Steady-state thermal-hydraulics calculations showed an important side of the improved safety of this concept. However, no transient or accident scenario calculations have yet been carried out, and the particular characteristics of the SHARP indicate the possibility of even more significantly improved parameters under transient or accident situations.

Fuel performance can be further improved with the use of new fuel designs (like hollow fuel pins) which will see their improved properties upgraded by the better thermal conditions of the SHARP's. A new parameter should be introduced in the neutronic and economic studies of the SHARP: this parameter is fuel enrichment. It has been shown(3) that increased fuel enrichments (up to about 5% U-235 enrichment) result in improved fuel utilization. This fact, theoretically verified for standard power level cores, can be synergistically potentiated with the demonstrated better burnup capabilities of a reduced-power core. In the economic field, this will impact in the conflicting factors of reduced refueling outages and increased fuel interest costs, which will require new evaluation and optimization.

The SHARP concept must also be tested for its advantages in a fuel reprocessing policy environment. More involved isotopic accounting methods will be required, but the favorable heat transfer situation of the SHARP's allows wide possibilities of play with core design variables that should certainly result in significant improvement of fuel cycle characteristics under a reprocessing policy environment.

It is also necessary to make a thorough evaluation of the possible monetary advantages obtainable from the improved safety aspects of the SHARP. These monetary advantages can come from reduced safety systems requirements, shortened licensing procedures,etc. Although apparently more speculative in nature, these aspects can result in significant savings, affecting the capital cost of the plant, and thus, the highest portion of the power generation cost.

APPENDIX A
METHODS OF IMPROVING BURNUP IN PWR'S

A.1. Introduction

A.1.1. Motivation and Constraints.

LWR nuclear power plants were rapidly deployed in the U.S. because of their significant advantages in known technology over all other types of reactors when the industrial application of nuclear energy began. The existence of the enrichment facilities and the Nuclear Navy program were key to the selection of the LWR's as the main U.S. reactor concept for deployment purposes. However, LWR's are obviously not the most efficient users of uranium resources. The "burners" were needed for rapid deployment and, with the LMFBR and reprocessing in mind, the LWR fuel efficiency was not a major concern. The fact that the discharged fuel contains large amounts of U-235 and plutonium was important from the point of view of resource utilization, with reprocessing and the breeder "around the corner".

However, in the present political situation in the U.S. where reprocessing has been indefinitely delayed and the development of the LMFBR is questionable, the fuel utilization problem requires a whole new view and approach. The new goal is now to obtain the maximum energy generation with the minimum ore consumption and with the lowest enrichment and fuel fabrication costs. Some of these

goals are at times in conflict, and a compromise (often just an optimum economic solution) must be reached by an adequate combination of these goals and the operational characteristics of the reactors.

Increasing the energy obtainable from a certain amount of ore may involve a very wide range of actions, from just operational policy changes to some reactor redesign. Some actions which in their conceptual sense might be in the right direction towards reducing fuel cost and/or ore requirements may actually be forbidden by safety or operational constraints. Some of the main restrictions to bear in mind while searching for methods of improving burnup are described below:

a). Linear power in the fuel pins is normally kept at a very high level. Any attempt to increase it must insure that adequate heat transfer at the pins' surfaces will be maintained, in order to avoid damages due to excessive temperatures or thermal shock. In general, any modification tending to increase the average linear power or the power peaking factor will need very special care.

b). Fuel pin structural materials and the fuel itself suffer physical degradation due to the radiation damage induced by the high fluences that they must withstand. Fuel manufacturers are trying to obtain fuel designs that will be able to withstand higher fluences than the ones presently allowed (goals are set for about 45 GWd/MTU average burnup to be obtained in the next 10 to 16 years) (2). Hence, it is not reasonable to contemplate a solution that would require fuel exposures much larger than the present

ones, if the solution is expected to be applicable in the near future. An average burnup of about 40 GWd/MTU would look reasonable as an upper limit in an optimistic scope for the near future.

c). Somewhat in connection with point a) mentioned above is the fact that if significant changes are made to the core lattice geometry, it is indispensable to assure heat transfer conditions consistent with operational and safety requirements of the core. For example, a reduction of pin diameter entices a reduction of linear power since the pin heat transfer surface would be reduced as the diameter is reduced.

d). The fully loaded core can not have an extremely high excess reactivity at BOL because this would force a high soluble boron concentration in the coolant, and this might cause a positive moderator temperature coefficient (MTC), which is unacceptable from a safety point of view. One way of avoiding the problems caused by a large excess reactivity at BOL is the use of burnable poisons, which also serve the purpose of reducing power peaking.

A.1.2. Schemes for Improving Burnup.

There are a number of schemes that can be implemented to increase the burnup obtained from a certain load of fuel. This section reviews the main schemes involved in extending burnup, while Section A.2 describes the possible practical ways of implementing these principles in a PWR power plant and goes deeper into the effects involved in the necessary modifications.

The core life terminates at end-of-life (EOL) because of two basic reasons: neutron economics (neutrons are lost by leakage, parasitic absorption, or both) and lack of fissile material (the fissile material is burned out and any bred fissile material is insufficient after awhile to outweigh the neutron losses). Any design scheme aimed at extending burnup must modify these conditions which affect the reactor, especially at EOL.

a). Reducing parasitic capture. A nuclear reactor has a number of materials that absorb neutrons preventing them from causing fissions in the fissile material. These are structural materials (cladding, grids, barrel, etc.), poisons (both control rods or residuals from burnable poisons), fission products (produced as the fuel is burned), the fuel itself and the coolant (either borated or unborated, the coolant presents some neutron absorption). A basic scheme to increase burnup could involve the reduction of the parasitic absorptions in the core. This can be done in different ways, although not all of them may necessarily be applicable in a real reactor, since relative burnup improvement must be weighed against any required redesign and secondary effects.

Parasitic absorption by fission products may be reduced, obviously, by reducing the amount of fission products present at EOL. This can be done by shortening the reactor cycle life and reducing the reload batch size. This way, at EOL there is more "low burned" fuel, and therefore less absorbing fission products. Another way of achieving the reduction of fission products in the

newer batches is the use of burnable poisons that will mostly disappear at EOL and will hold the power of the new batch down, preventing the development of large amounts of fission products in that batch. Burnable poisons should ideally be imbedded in the fuel material itself rather than in poison rods. In this way they can achieve a more thorough burn by EOL, and they don't have the absorbing structural material associated with the poison rods.

Parasitic absorption in the fuel itself can be reduced by reducing the power level of the reactor. This would reduce the fuel temperature, and therefore the Doppler broadening of absorption resonances. If some more effective means of cooling the fuel could be devised, the same effect would be achievable without the need for reducing the core power level.

Absorptions in the structural materials of the core could be reduced by replacing stainless steel parts with parts manufactured with less absorbing materials, and/or reducing the amounts of structural material present in the core.

Absorptions in the coolant are difficult to reduce at EOL, since no boron is present in the water. However, from the sole point of view of absorptions in water at any point in the cycle life, they can still be reduced by using lattices with less water or by using the water-heavy water mixture whose other use and side consequences will be described later in this section.

b). Reducing neutron leakage. There is a small proportion of neutrons that leak out of the core and are lost for fissioning purposes. There are some ways in which this leakage can be

reduced. One way is to place older fuel at the periphery of the reactor, instead of the presently current practice of having the fresh batch placed at the core periphery. This is further discussed in section A.2.3. Another way would be to improve the reflecting characteristics of the reflector, for example by changing the material that forms the core baffle.

c). Improving moderation. For safety reasons, PWR's employ a water/fuel ratio which is below the optimum for moderation. At current EOL's, the core could still be critical if the water/fuel ratio was closer to the optimum. Assuming the necessary safety margins could be assured, alterations of fuel lattice pitch or fuel pin diameter could improve the moderating characteristics of the system. A similar effect can be achieved by altering the water density, which can be done without need for fixed alterations of the system.

d). Improving conversion ratio. Since at EOL a large part of the power of the reactor is generated by the plutonium that has been bred from absorptions in U-238, increasing the conversion ratio would produce more plutonium, and therefore would tend to extend the core life. There are two ways in which this can be achieved in a PWR: The first is to employ the "spectral-shift controlled reactor" (further described in section A.2.9), which involves the mixing of light water and heavy water in a proportion that can be optimized throughout the core life. Since heavy water does not have the high moderating power of light water or the higher neutron absorption, a harder spectrum is obtained in the

reactor when the heavy water proportion of the coolant is increased, and a higher conversion ratio is possible. Another way is to have a variable pitch, which may allow for different water/fuel ratios at different times in life or in different regions of the core.

A.2. Techniques for Improvement of Burnup

A.2.1. Increasing Number of Batches.

In most PWR's, fuel management is based on adding or withdrawing the fuel elements in batches, in such a way that all the elements belonging to a certain batch have at any time in life a similar burnup level.

Some simplified methods of calculating burnup capabilities of PWR's use the assumption that all batches in the core share the same fraction of the total core power (provided they have the same number of fuel assemblies, which is normally the case) and that the core multiplication factor can be obtained as the average of all the batches' multiplication factors (28). This method was used, for example, by L.E. Strawbridge of Westinghouse. Under this assumptions it is possible to determine that the burnup at discharge of a fuel batch, Xd, can be calculated as:

$$Xd = Xc * (2*N / (N+1))$$

where Xc = Average burnup of the core at EOC.

N = Number of batches present in the core.

An analysis of the extreme values of N yields an appreciation of the available choices. N=1 corresponds to a reactor where the whole core loading is renewed at each refueling, and it is obvious that in this case the discharge burnup is the same as the core average burnup at EOL. For N very large, Xd tends to be equal to $2*Xc$, i.e., the discharge burnup can be, in the limit, twice as large as the reactor average burnup at EOL. This would be the case of an on-line refueling reactor such as a CANDU or a MAGNOX. The increased burnup achievable, according to this theory, from the same initial fuel as the number of batches increases can be observed in Figure A.2.1.
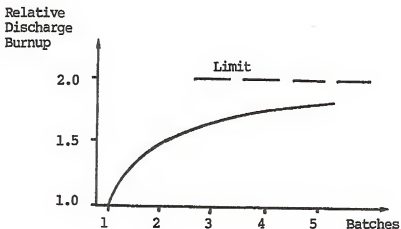

Figure A.2.1. Increase in Fuel Burnup due to Partial Refueling.

Although the assumptions used for this analysis are not quite true (i.e. the batches do not really share the same amount of power and the core K-eff. is not really the average of the batches' K-eff.'s) it is certainly true that increasing the number

of batches of a reactor will drive the fuel to higher burnups, for the same initial enrichment. On the other hand, if the enrichment is kept constant, having more batches in the core reduces the size of the reload batch. Consequently, the time between refueling shutdowns is reduced, which is generally not desirable because it lowers plant availability.

The immediate solution to these "very-short" cycles is to increase fuel enrichment so that a higher core average burnup can be achieved at EOL, and thus, longer times can elapse between consecutive refuelings. R.L. Hellens showed (3) that in addition to improving the discharge burnup, increasing the number of batches and fuel enrichment also improves ore utilization for enrichments up to about 5% U-235, as can be observed on Figure A.2.2.

In any case, a 5% enriched fuel used in a 4- or 5-batch core would theoretically drive the fuel to burnup levels far beyond the present technological possibilities. If fuel pins were made which could stand average fuel burnups of about 50 GWd/MTU, potential ore savings of up to 15% could be achieved. However, with the present state-of-the-art fuel and the assumptions stated in Section A.1, it is feasible to obtain 40 GWd/MTU with a 4-batch core, a 12-month cycle and about 4% enriched fuel, with an associated improvement in ore utilization of about 10% .

The main problem associated with such a maneuver, as described above, is that increasing the number of batches in a core containing a constant number of fuel assembly locations increases the probability of placing highly burned assemblies adjacent to
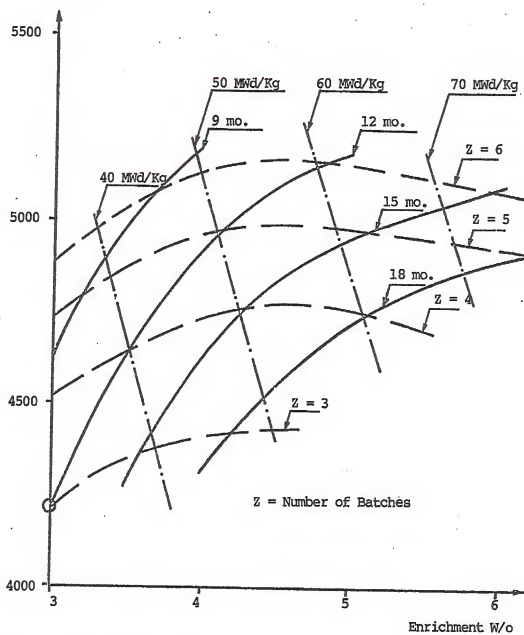
Figure A.2.2. Ore Utilization vs. Enrichment and Cycle Length.

fresh and relatively highly enriched assemblies. This represents a serious problem for in-core fuel management, since this induces severe power peaking problems which are difficult to negate. Burnable poisons utilization is then necessary, which further complicates core management. Burnable poisons and their associated advantages and disadvantages are described below.

A.2.2. Burnable Poisons.

Burnable poisons accomplish several functions simultaneously: reduction of core reactivity at BOL, allowing a reduced chemical shim and thus make a positive MTC less likely; reduction of power peaking from one assembly to another; power shaping within a fuel assembly; and power reduction on the fresh assemblies, which stretches the cycle length. Each of these aspects is examined separately below.

There are two ways of implementing burnable poisons in PWR's. The first and only one used currently in PWR's employs the so-called "Lumped Burnable Poisons" (LBP's) also called Burnable Poison Rods (BPR's). They consist basically of rods having nearly the same geometry as a fuel pin and they are loaded with a poison (often borated Pyrex) with concentrations of boron such that it may be completely burned in one reactor cycle life. The rods are placed inside a thimble so that they can be easily removed at the end of the first cycle, leaving the fuel assembly unpoisoned for the following cycles. This type of poison does not represent a very difficult problem for core calculations. Although the calculations normally performed to determine the effective cross

sections as they change through the cycle and the isotopical burnup evolution of the LBP involve some trial-and-error adjustments, they can be treated rather easily. However, they present some physical problems:

a). The absorption cross section is not close to zero at EOL because of all the structural material associated with the BPR and because some poison atoms are always left. This causes a residual negative reactivity which is undesirable at EOL, since it causes the cycle to be shortened.

b). When the BPR's are removed for later cycles, the channel that housed them is filled with moderator and mildly absorbing flow-control devices, forming a sort of flux trap, and causing the neighboring pins to have a higher-than-normal power. In other words, the assembly is left with a power distortion.

c). The array spaces used for BPR's can never be used for fuel pins. The poison rods are removed at the end of the first cycle of core residence, but the guide tubes remain in their places, and there is no reasonable way of removing them and placing fuel pins in these locations. Even if such a manipulation were possible, severe local power peaking problems would appear because of the drastic differences of burnup status among neighboring pins. LBP's therefore represent a loss of energy that could have been produced by the fuel pins that were displaced.

The second way of implementing the burnable poisons is to load the fuel pellets with Gadolinium. This technique is used in BWR's

but has not been used yet in PWR's as far as the author knows, although some tests are being conducted.

The reactivity worth of gadolinium-loaded fuel pins and its evolution through core life are much harder to calculate accurately than the BPR's and they also require increases in fuel fabrication cost, because more types of fuel are needed and more complex and careful control is required during manufacture. On the other hand, Gadolinium-loaded pins present some substantial potential advantages such as:

a). Complete burning of the poison by EOL is more likely because of the poison being mixed with the fuel itself.

b). No residual absorption due to extra structural materials is present.

c). No fuel loading is lost, since Gadolinium is imbedded in the fuel pellets.

d). No power shape distortions due to water holes that are present in following cycles.

Gadolinium-loaded fuel pins seem to have a promising future in PWR applications. A review all the functions accomplished by the burnable poisons follows.

a). Since burnable poisons are added to the most reactive fuel elements or to the regions where power would peak, their presence causes a strong reduction of the core reactivity. The effect of the poisons is maximum at BOL, since they have not been burned yet. This effectively reduces the amount of chemical shim needed for criticality of the core, which is an interesting effect,

since the high soluble boron concentration needed at BOL may cause a positive MTC, with intrinsic safety implications.

b). Since they are placed in the fuel assemblies having the highest power levels, the burnable poisons reduce the power peaks between assemblies and thus allow the whole-core power level to be increased while preserving safe heat transfer conditions in the most restrictive assemblies. The end result is a higher power output by the reactor.

c). For assemblies having very assymetrical boundaries (e.g. a fresh assembly having the baffle and reflector on one side and another fresh assembly on the opposite side), burnable poisons may be placed in such a way that they reduce the power tilt that would appear within the assembly itself. This avoids severe power peaking problems in subsequent cycles, when the assembly does not have burnable poisons anymore. An example of such use of LBP's is shown in Figure A.2.3 (37).

d). Since their reactivity and burnup are held down during the whole first cycle, the poison-loaded fresh assemblies show a higher reactivity at EOL, which allows the core to stretch the burnup obtainable in each cycle, and with this, the discharge burnup obtainable from the same initial fuel. This can be seen rather easily using Strawbridge's rough approximation which states that for a core having several batches of the same initial enrichment (i.e. for an equilibrium-cycle core), EOL will be reached when the core average burnup reaches a certain constant
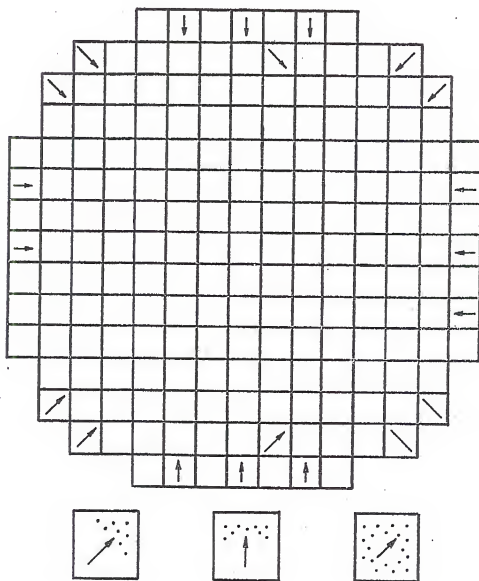
Figure A.2.3. Orientations of Non-symmetric Burnable Poisons.

value, determined by the enrichment of the fuel and the in-core fuel management scheme.

With this assumption, the core-average burnup at EOL can be calculated (for a three-batch core, for example) as:

$$CA = \frac{CB*P1 + CB*(P1+P2) + CB*3}{3} = CB * \frac{2*P1 + P2 + 3}{3}$$

where CA = Core Average Burnup at EOL.

CB = Cycle Burnup.

P1 = Relative power of Batch 1 (in the core for the first cycle).

P2 = Relative power of Batch 2 (in the core for the second cycle).

Then, if for a given enrichment and fuel management scheme CA is fixed, it is possible to obtain CB as

$$CB = \frac{CA * 3}{2*P1 + P2 + 3}$$

Since the discharge burnup for the fuel will be three times the cycle burnup, it is obvious that the lower P1, the higher the discharge burnup can be, and P2 has the same effect, to a lesser degree. These considerations indicate that the burnup of the fuel should be shifted towards the end of its life as far as it is possible.

It is important to bear in mind that the assumptions used for these calculations are not exact, and it is obviously not reasonable to rely too heavily on the precise numerical results

obtained from them. However, the trends shown are close to reality, and thus it is possible to state that the use of burnable poisons increases the discharge burnup (and therefore ore utilization) by reducing the fresh batch's relative power.

A.2.3. Low-Leakage Fuel Management.

Most LWR's operate on a multiple-batch fuel management scheme because of the increased burnup obtainable (as shown in Section A.2.1) and the associated possibility of power flattening. It is also most common in large PWR's to use a 3-batch scheme, where the fresh batch is loaded at the core periphery, and the once-burned and twice-burned batches are placed in the inner region of the core in a "checkerboard" pattern. One such typical disposition is shown in Figure A.2.4 (26).

This disposition of the "second cycle" batch being closely mixed with the "third cycle" batch is used to drive the latter, which in turn is used to avoid power peaking in the former; this peaking can occur because of the relatively high reactivity of the "second cycle" batch and its location in a high-worth area in the core. The "first cycle" high reactivity batch at the core periphery raises the power in that zone, which is a low worth region; this same effect helps maintain the batch's power at a reasonable level. However, Figure A.2.3 is an example which shows that even in the peripheral "low worth" situation, some "first cycle" fuel elements may need LBP's in order to prevent their power from being too high.
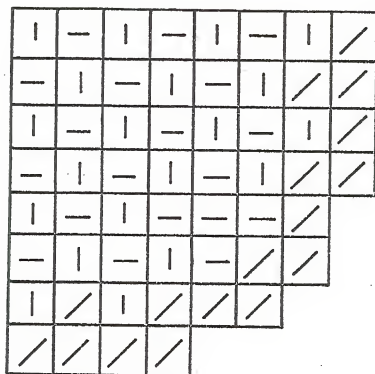
Figure A.2.4. Batch Distribution in Core.

It is reasonable that such an in-core fuel management scheme, which uses the close mixing of different-burnup fuel assemblies and the different worth of the regions in the core, obtains a flat or near-flat power distribution without much complication, and avoids high power peaking factors. However, this scheme is rather poor when one is concerned with neutron economy. The "freshest" batch lies on the periphery, surrounded by a steel barrel and a moderately absorbing refelector (depending on the chemical shim), and its high neutron production is offset by a disproportionate neutron loss in these adjacent materials. This is no great problem at BOL because there is excess reactivity just about everywhere, but unproductive and disproportionate neutron losses at EOL cause the reactor to go sub-critical earlier than otherwise possible.

A new "low-leakage" fuel management scheme has been developed, where neutron leakage is minimized at all times, thus trying to stretch cycle life by simple neutron economy. In this low-leakage scheme, the fresh batch is placed in the inner region of the core, "checkerboarded" with the "third cycle" batch, while the "second cycle" batch is put on the peripheral region. Consequently, this fuel management scheme is called "in-out-in" to describe the succeeding positions of the batch through its life, as opposed to "out-in-in" which describes the placement of the fuel in the currently used scheme.

With this in-out-in scheme, the "third cycle" batch is driven by the fresh batch. This allows the oldest batch to be driven to higher burnup at the same time that the fresh batch is more

effectively used by the close vicinity of the highly burned assemblies, and its neutrons are not lost in the baffle or reflector, but are utilized in an old batch for further fissions or conversion. The "second cycle" batch being in the core periphery yields a smaller neutron leakage than a "first cycle" would (particularly important at EOL) because its pins have already been burned for two complete cycles by the time EOL is reached. Obviously, this "second cycle" batch does not generally need LBP's at all.

This in-out-in scheme is better than the out-in-in scheme as far as neutron economy is concerned, and is able to improve burnup by about 3% (2). However, it presents the problem of high power peakings. This problem is aggravated by the fact that power peaking does not improve as the cycle advances (as was the case with the out-in-in scheme) but it gets worse. This is due to the fact that the fresh batch burns its poisons (which are obviously necessary) and appears relatively clean and low-burned at EOL while facing a crisp-burned "third cycle" batch. In addition to being an operational problem, this is also a calculational problem, because the whole cycle burnup must be followed on a pin-by-pin basis before the power peaking constraints may be obtined, since they appear at EOL.

To the best knowledge of the author, there is yet no commercial power reactor using this kind of fuel management, but it is being given serious study and consideration. It appears as if Gadolinium-loaded pins might be of considerable help in reducing

the power peaking problem associated with the in-out-in scheme. However, this would entail, as mentioned in Section A.2.2, yet a further calculational, technological and economic problem.

A.2.4. Alterations of Fuel-to-Water Ratio.

Changing the relative amounts of water and fuel in the core can have a significant impact in the nuclear characteristics of the reactor. When more water is present, the neutron energy spectrum is softened and the multiplication factor is raised (since PWR's work with an undermoderated configuration for safety reasons). On the other hand, the epithermal neutron population is then decreased, and captures in U-238 are reduced, thus reducing the conversion ratio and limiting the amount of plutonium contributing to stretch the cycle length. The reverse effects appear when the water proportion is decreased.

The two effects described are competing from the point of view of cycle length; however, considering the present design of PWR's, burnup can be increased by going to a more moderated configuration. There are four possible ways of changing the fuel-to-water ratio:

    a). Change the pin diameter.

    b). Change the lattice pitch.

    c). Change the effective fuel density.

Changing the pin diameter has the advantage of needing a very limited redesign, since it is possible to keep the same basic fuel assembly design and backfit the change into operating reactors. As a matter of fact, Westinghouse Corporation has a new optimized pin of reduced radius that will be slowly introduced into the market

and will eventually be installed in reactors that are now using the larger pin design assemblies.

Changing the lattice pitch avoids the problems associated with heat transfer, but makes backfit into current LWR's much more difficult since it could call for major redesign of the reactor core unless the pitch variation were very small.

Changing the fuel density can be done in several ways: reducing the fuel density and filling the "empty" space with some low cross section material; or just reducing the physical fuel density; or using annular fuel pellets. In any case, the fissile inventory can be maintained by increasing the U-235 enrichment.

Combustion Engineering did a study (3) where fuel density was changed, while keeping the total U-235 inventory constant. Other alterations were also examined, both separately and in combination with the fuel density alterations. If the fuel density is kept at the present value, ore savings of about 6% can be achieved by going to a larger pitch. Larger savings can be theoretically achieved (up to about 10%) by reducing the fuel density to about 60% of the present value while keeping the same pitch, but this would drive the fuel to burnups far beyond the acceptable limits. However, a combined variation in which the fuel density is reduced to about 80% of nominal, and the lattice pitch is increased could allow the ore utilization to be improved by about 8% while keeping the exposures within acceptable limits. From the core cycle point of view, this would correspond to a rather high burnup combined with a
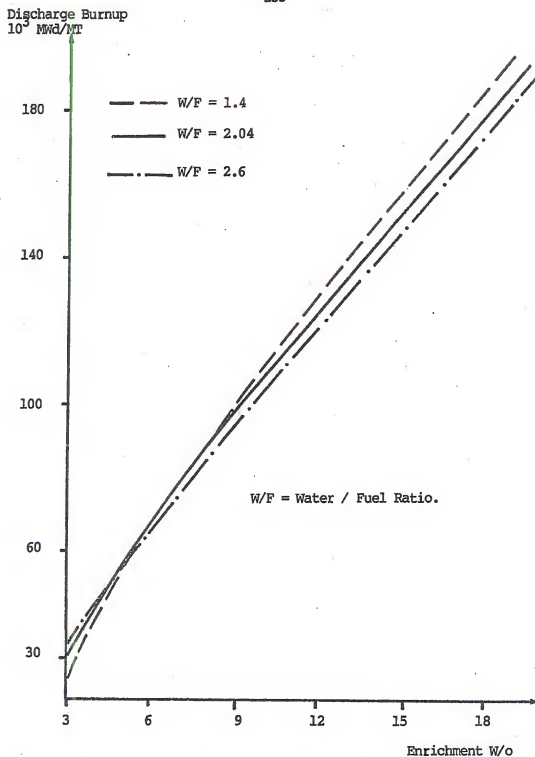
Figure A.2.5. Discharge Burnup vs. Enrichment at Several Fuel Densities.
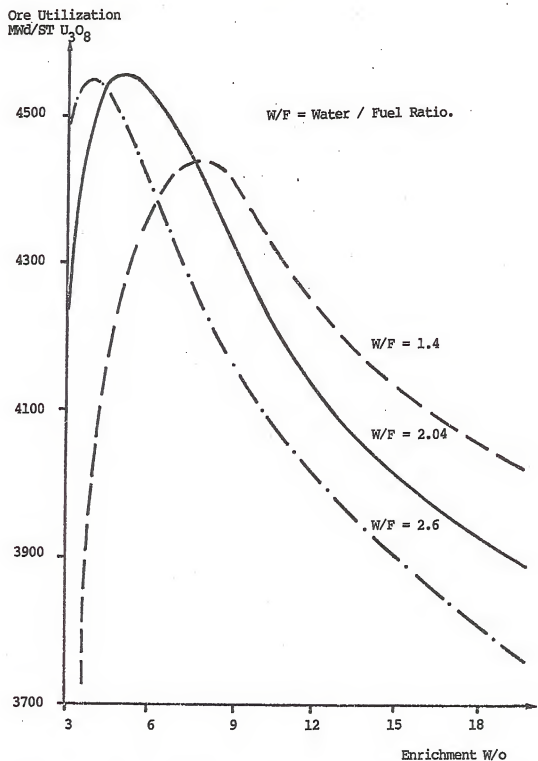
Figure A.2.6. Ore Utilization vs. Enrichment at Several Fuel Densities.

long cycle length. Figures A.2.5 and A.2.6 show the results of the study performed by Combustion Engineering.

A.2.5. Low Power Density.

Reducing the power density of the core is another means of improving burnup. It also has the very desirable side effect of greatly increasing the safety, relieving the general operating conditions and hence reducing the problems of operating the reactor.

Reducing the power density of the reactor does not necessarily imply a reduction of the coolant temperature which would result in a reduced thermodynamic plant efficiency. In the reduced power density cores, the coolant is kept at the standard operating temperature by altering the flow conditions, and thus, the plant efficiency remains unaltered. The increase in safety of the reactor is due to the reduced surface heat flux (which makes reaching critical heat transfer conditions more unlikely) and to the reduced heat storage and temperature of the fuel.

There are two main factors contributing to the burnup extension of a low power density core: The reduced Doppler effect and the reduced Xenon concentration. This modification can contribute significantly to the relief of several thermalhydraulic and operational problems, while presenting no new technological obstacles. However, it has the disadvantage of not being retrofittable (unless current plants were to under-use much of their balance-of-plant equipment) and of requiring a higher fuel inventory per installed power. On the other hand, it presents a

great improvement in plant operational flexibility, and may allow for longer core cycles, reduced outage time and reduced personnel radiation exposure, which are very desirable features.

Westinghouse Corporation conducted research on this subject (2) for a reactor whose power density was reduced to 75% of the current standard value. A possible ore savings of 3% were obtained, although the extra costs associated with the rest of the fuel cycle placed the option at the same fuel cycle cost level as the standard reactors.

The author performed some calculations for reactors in which power densities were reduced to 50, 33 and 25% of the standard values and obtained burnup increases of 3 to 4% with respect to the standard reactor, depending on the power density chosen. Allowing the pitch to vary and going to better-moderated configurations increased the burnup improvements to levels between 5.5 and 7% above the standard core, depending again on the power density chosen.

It appears that most of the burnup improvement achievable is obtained when cutting the power from 100% to 50%; little improvement is obtained for further reductions to 33 or 25% power density levels. Table A.2.1 shows the discharge burnups obtained for the various power levels and their percent increase over the standard core burnup.

Table A.2.1. Discharge Burnups of Low power Density Cores.

Power Level	Discharge Burnup (MWd/MTU)		% burnup increase	
%	Std. Pitch	Optim. Pitch	Std. Ptch.	Opt. Ptch.
100	36183	——	0.0	——
50	37296	38134	3.1	5.4
33	37359	38397	3.3	6.1
25	37634	38541	4.0	6.5

A.2.6. Flattening Axial Power Distribution.

If all the pellets in a PWR had the same composition at a given point in time, and no special absorbing materials were present, the reactor power distribution will approximate a cosine in the axial direction, which means that each pin would be burning much faster around the reactor mid-plane than towards the top or the bottom. Since the reactor must be designed for the most unfavorable spot to work safely, this means that most of the core is under-employed with regard to power generation and fuel utilization. It is, therefore, highly desirable to alter the "natural" axial cosine shape of the reactor power distribution and make it flatter, so that power peaking is reduced and fuel average discharge burnup is increased by a more uniform burnup.

Such capability is available in present reactors, although it might be possible to develop improvements. One available axial power flattening method currently in use employs partial-length

neutron absorbing rods. They are distributed throughout the core cross section and then normally placed at about mid-height in the axial direction in order to reduce, as far as possible, the power peak that normally appears at the core midplane. The main problem associated with the partial-length control rods is that they take the space of fuel pins or full-length control rods, and it is just not possible to deploy many of them; this causes the axial power flattening to be imperfect. Figure A.2.7 shows the effect of a partial-length rod on the axial power distribution of the Oconee-1 reactor (38).

It is hard to devise an element that could do a similar job without occupying fuel pin or full-length control rod spaces because partial length rods need to be movable. This is true for several reasons: one is that they may be used to help control reactivity transients, like xenon oscillations; another is the fact that if an absorbing material is placed in a fuel assembly in a non-movable manner, it causes a reduction of the core life because it absorbs some of the highest worth neutrons in the reactor, which is not desirable at EOL.

A.2.7. Increasing Enrichment.

If the enrichment of the reload batch of a reactor is increased, it seems obvious that the core should be able to achieve higher burnup, causing the discharge burnup to be also higher. If the fuel management scheme is kept constant and the same batch size is used, achieving a higher cycle burnup would entail longer cycles, which is a desirable trend from the economic point of view.
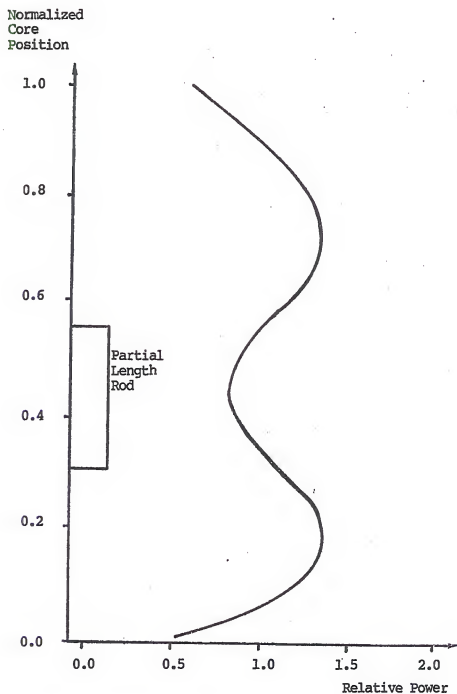
Figure A.2.7. Power Shaping Effect of Partial Length Rod.

However, a higher enriched fuel requires larger quantities of ore for each core reload. Although for very low enrichments an increased enrichment also causes a better ore utilization, a maximum is soon reached after which further enrichment entails more ore consumption for the same energy produced, as can be observed in Figures A.2.2 and A.2.8. In addition to this, the combination of higher enriched feed fuel and further burned old batches will tend to cause more severe power peaking problems, making burnable poisons probably necessary.

Combustion Engineering research on this topic (3) shows that with a 3-batch out-in-in fuel management scheme (which is the most commonly used in large PWR's), ore utilization could be theoretically improved by about 6% by increasing fuel enrichment to about 4.8% U-235. However, this would require the fuel to be driven to over 50 GWd/MTU, which is beyond the present technological limits.

According to the limits set in Section A.1, ore utilization can be improved by about 3 to 4% by going to fuel enrichments of about 3.8% U-235. Figure A.2.8 illustrates the ore utilization improvement achievable for different values of fuel enrichment and their associated discharge burnups assuming a 3-batch out-in-in fuel management scheme.

A.2.8. End of Cycle Coastdown.

Power coastdown at EOC is an operation which is often used by utilities to meet shutdown schedules rather than for burnup or resource utilization improvement. There are two ways of performing
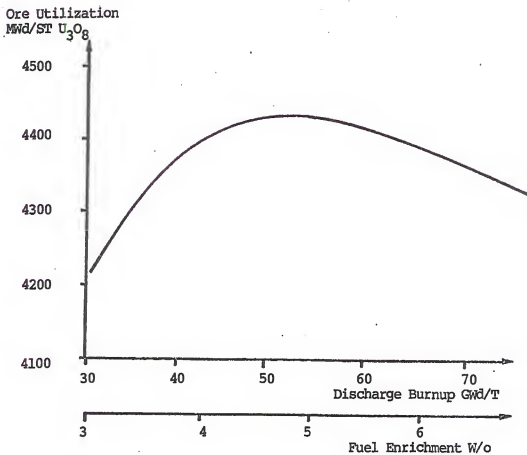
Figure A.2.8. Ore Utilization vs. Fuel Enrichment and Burnup.

the EOC coastdown: reducing the coolant temperature while keeping the full thermal power of the reactor, or reducing both the coolant temperature and the core power. The first system allows for increased burnup, but practically no ore savings is obtained at all as shown by Westinghouse's research (2) on this subject. The second system may allow a 7% savings in ore cost and a 4% savings in fuel cycle cost. However, the total generation cost may not be improved at all, or at best may be improved to some limited extent; this is a consequence of the extra costs of the replacement power for all the period during which the plant is delivering a reduced power level.

Coastdown must be carefully planned, because driving the core to a limit will shorten the length of the next cycle. EOC power coastdown appears as a feasible operational adjustment, but its overall economy is not quite clear for a systematic use.

A.2.9. Other Possibilities of Minor Importance.

The options discussed in the previous sections appear to be the best present candidates if a policy of increasing ore utilization or extending burnup is to be seriously undertaken in the near future. There are, however, many other ideas which might in some way help increase burnup, but which either represent a very small potential effect or their implementation would be difficult and/or expensive. Some of these less possible ideas are discussed below.

a). Zircaloy baffle. Present reactors use a steel baffle, which presents a considerable absorption cross section. Zircaloy

could perform the same functions, presenting better nuclear characteristics. Backfit of this idea would be really expensive, while potential ore savings are very small.

b). Forcing a "migrating" axial power peak. By means of partial length rods, it might be possible to force the axial power peak to travel along the reactor axis, forcing a more leveled burnup towards the extremes of the fuel. Although some extension in burnup might be expected, it is possible that rather strange power distributions would appear. The concept would require a really sophisticated system of partial length rods.

c). Reversible-halves fuel assemblies. Since the axial power peak is towards the center of the reactor while the top and bottom ends have a lower power, one way of obtaining a more uniform discharge burnup along the whole axis length would be to use fuel assemblies cut in two halves. At the third cycle, for example, the two halves could be turned upside down, causing the center zone to be at the extremes, and vice versa. During the third cycle, the assemblies would receive the maximum burn at a region that had the lowest exposure previously. Of course, this would increase the fuel fabrication cost in a significant proportion. It would also increase the refueling outage time because of the higher complexity of operations required, and probably most important of all, it would drastically worsen the power peaking problem.

d). Another way of flattening the power distribution could be through axial enrichment zoning of the fuel. This would reduce the

LEOPARD (11) is a fuel cell code intended to calculate neutron flux spectra and isotope cross sections as they change with fuel depletion. It is based on modified versions of the MUFT (12) and SOFOCATE (13) codes. The code performs spectrum calculations and fuel burnup calculations, assuming the fuel is imbedded in an infinite array of elementary cells. The code provides the possibility of specifying the presence of materials that cause alterations of neutron spectrum, but which are not related to the elementary cell itself. Such is the case of instrument thimbles, water holes, etc.

Although the spectrum calculations take into account the geometry of the fuel cell, the burnup routines are entirely non-dimensional and fuel depletion is considered uniform throughout the pellet region. The code accounts for the thermal expansions of the different materials in the cell, according to the temperatures supplied by the user. Input to the code is reduced and relatively simple. However, LEOPARD has the following drawbacks:

a). The power level of the fuel can not be changed from one timestep to the next, and therefore, the fuel is burned at a constant power level which results in a false equivalence of time and burnup level for each particular case.

b). The calculation of resonance self-shielding is allowed only for U-238. which is particularly inaccurate for high levels of burnup where other isotopes (such as Pu-239 or Pu-240) have significant concentrations.

On the other hand, the code is extremely flexible, allows the variation of poison levels during life. and runs in moderate times: A total fuel burnup calculation can be run in just a few minutes of computer time. The most significant data input by the user are the following:

a). Geometrical data describing the cell and the "extra" region.

b). Composition specification for each material zone.

c). Temperatures for each region. densities. pressure, and perpendicular buckling.

d). Flags indicating units of input data. type of cell geometry (square or hexagonal). number of broad groups to be used for the output cross section tables. option for buckling search, etc.

The code prints a report for each timestep specified during the life of the core. The main items listed on each report are the following:

a). Cell-homogenized number densities of each isotope at the beginning of the timestep. and conversion achieved during the timestep.

b). Microscopic cross sections of all the isotopes present in the cell. and macroscopic cross sections of the smeared cell. All

cross section tables are produced for the number of energy groups specified by the user.

c). Fast and thermal fluxes, average neutron velocity, group-wise and total multiplication factors, etc.

The EPRI-CELL (19) code computes the space, energy and burnup dependence of the neutron spectrum and isotopic cross sections within cylindrical cells of light water reactor fuel rods. Its primary output consists of broad-group, microscopic, exposure dependent cross sections for subsequent use in multidimensional neutron diffusion theory depletion analysis.

The code is an answer to the need for a well standardized and proven code able to supply the sets of burnup and/or nuclide concentration-dependent cross sections for use in the large multigroup, multidimensional, diffusion theory codes used in the nuclear industry for fuel management calculations and core following. This requires the following characteristics of the code:

a). The code must be based on well-proven previous computational schemes and must take into account all the parameters that have been shown to have a significant effect on the fuel characteristics as it undergoes the fission process.

b). The code must be flexible enough to allow representation of all the types of fuel situations that can be normally encountered when performing LWR calculations, yet it has to be as simple as possible to the user. This is solved by providing two

input options, one of which is called the "engineering" input option, and defaults most of the commonly used data, while the other option, called the "general" input option, requires full specification of the problem.

c). The code must provide an output which is capable of being easily processed for input to the core model codes.

The result is a code with an extremely simplified input, but nonetheless, with a large amount of options available. It is based on the very well-known codes THERMOS (21), GAM (20) and CINDER (22). It simulates the burning of a LWR fuel pin in which the geometry and compositions are specified by the user, as well as the circumstancial data concerning the burnup process, such as power level variation through time, soluble boron concentration, etc. The output includes a printed listing of microscopic isotope cross sections, macroscopic cell-averaged cross sections, isotopic number densities, multiplication factor, etc. and a magnetic device (tape or disk) data set with the same type of data in an adequate format to be transformed into cross section tables for the core model codes.

Some of the main characteristics of EPRI-CELL are the following:

a). The code includes the effects of Dancoff factors, correcting also for the fuel rods that are not surrounded by other fuel rods but that are adjacent to interassembly water gaps, to water holes, instrument thimbles, etc.

b). It also includes the spectral effects of the structural materials that are present in the reactor but that can not be assigned to the geometry of the elementary fuel cell.

c). The code accounts for the resonance energy shielding of all the nuclides present in the cell.

d). The thermal energy cutoff used is 1.855 eV, which is more adequate than other codes' lower values.

e). Some calculational and convergence techniques of the "parent" codes have been modified to improve their accuracy and/or to optimize convergence.

f). The isotopic neutron cross sections are computed from the microgroup libraries of GAM and THERMOS, with 62 and 35 energy groups respectively.

g). The depletion calculations are performed for each mesh space inside the fuel pellet, and isotope accounting is kept separate for each mesh region. Depletion is done with a four group spectrum and cross section data set.

The main items input by the user in the general input option are the following:

a). Definition of the geometry of the problem: Number of material zones, thickness of each zone and number of mesh points assigned to each zone.

b). Specification of the nuclides present in the cell (maximum of 25) and their number densities for each material composition.

c). Volumetric fraction and density fraction of each composition in each material zone.

d). Number of timesteps desired for the total calculation; duration of each timestep, power level associated with each timestep and soluble boron concentration.

e). Option flags determining buckling search, correction for resonance overlap, presence of heavy scatterer, boundary conditions, type of isotopic library data, number of broad groups used for collapsing and editing (minimum of 2, maximum of 5), type of data table output, etc.

f). Convergence criteria, temperatures for each material zone, type of fission spectrum, optional group-dependent buckling specification, information about nuclides for which resonance calculations are requested, etc.

The input for the engineering option assumes default values for many of the variables used by the code. The input is reduced to a minimum while keeping a fair flexibility, and still allowing representation of most of the classical problems in fuel burnup calculations. The main items input in the engineering option are the following:

a). Weight fractions of fissile materials in the fuel pellet and density fraction of the pellet.

b). Volume fractions of structural materials in the buffer extra region surrounding the fuel cell.

c). Composition of the cladding material.

d). Number of total timesteps used in the calculation, soluble boron levels and power levels.

e). Geometrical data: radii of the zones of the fuel cell, pellet and clad inner and outer diameter, pitch, and extra region volumetric fraction.

f). Option flags concerning units used in input, buckling search, number of collapsed groups, type of edit and output data tables, etc.

g). Temperatures, spectrum specification and convergence criteria.

APPENDIX B6
NUPUNCHER CODE DESCRIPTION

NUPUNCHER (23) is a coupling code whose function is to prepare the cross section tables output by EPRI-CELL to the magnetic device into a format compatible with the HARMONY (24) tables needed for PDQ-7 (17) calculations. This results in a substantial simplification of the most cumbersome part of the input to PDQ-7.

Although NUPUNCHER's task is apparently simple, it contains a considerable amount of options due to the flexibility built into the HARMONY tables. The code generates the required macroscopic and microscopic cross section tables using the "burnup" variable as basic independent mask, but allows the user to specify any other number of masks for control of any particular set of microscopic cross sections. For example, it is common practice to assign the absorption cross section of Pu-240 to a table controlled by the isotope's own number density rather than assigning it to the general table, controlled by the burnup level of the fuel. NUPUNCHER accommodates any of these types of changes and generates the HARMONY tables accordingly. The main items input to the code by the user are the following:

a). Identification of the data set containing the EC-DATA file generated by EPRI-CELL and definition of the material names to be used in the HARMONY tables.

b). Table identification number, interpolation information, number of energy groups and EPRI-CELL timestep number to be used for non-interpolating data.

c). Table number for macroscopic tables, type of data to be used in the macroscopic tables, first mask number and first interpolating table number.

d). Definition of interpolating tables that must use a mask other than "burnup"; definition of the mask.

e). Option to punch PDQ-7 depletion chains.

PDQ-7 (17) is a multigroup, multidimensional diffusion theory code, used for the modelling and following of reactor cores. It is one of the most powerful calculational tools now in use in the nuclear industry. It has an extreme flexibility, and almost any neutronic situation where diffusion theory can be applied can be adequately represented with the code.

This great power and flexibility entails a long and complicated input specification, to the point that several codes have been developed (such as NUPUNCHER (23), described in APPENDIX B6, and CHIMP (36)) which assist the user by taking cross section data from codes like EPRI-CELL (19) and LEOPARD (11) and translate them into a format suitable for input to PDQ-7. These auxiliary codes still leave a significant task to the user, but they can take care of the most voluminous and cumbersome part of the input deck.

PDQ-7 has no cross section library of its own. Cross sections for all the isotopes must be provided from an external source. However, PDQ-7 has a cross section handling routine, HARMONY (24), which allows the code to evaluate the cross sections of all isotopes as a function of up to three independent variables, and make interpolations in order to provide the most accurate estimate of the cross sections for each situation of the core. The current

practice is to specify most of the cross sections as a function of the burnup level of the fuel. It is also possible, for example, to specify them as a function of both the burnup level and the soluble boron concentration; the code will make the necessary multi-dimensional interpolation for the particular situation of the isotope at the moment of cross section evaluation. Note that the cross sections are evaluated for each isotope and for each material zone in the core, so it is possible to have different cross sections for a given isotope at a given time, if the isotope is present in two different regions. In order to make this possible, the burnup level is computed and accounted for as a separate item for each diffusion mesh space. This is an indication of the extreme sophistication of the cross section handling procedures, but it suggests as well the fact that PDQ-7 requires enormous amounts of computer memory and time.

The code is also very generalized and flexible in the geometry specification. It accepts one to three dimensions, with the mesh spacing of either rectangular or hexagonal type in the X and Y directions. The mesh and material zone specifications are done in such a way that it is simple to build large structures with repetitive patterns, as is the case with most reactors.

PDQ-7 requires also the specification of the radioactive and transmutation chains with all the decay times involved in them. This represents a further task for the user (although it is usually taken care of by the auxilliary codes mentioned above) but it

enables the user to choose as simple or as complex of an isotope accounting procedure as may be needed for each particular job.

Another involved task left to the user is the specification of the groups of chains that must be assigned to each material region, the interpolating tables that correspond to each set of isotopes, and the respective masks (independent variables) that are controlling each of the interpolation tables. PDQ-7 can also admit "burnable" and "non-burnable" compositions, which obviously receive a different treatment. Non-burnable materials can also have interpolating tables, but their cross sections are stored in the macroscopic form, since there is no need for accounting of isotopic number densities.

Finally, the code requires a set of flag and options specifications, indicating the type of problem to be solved, the type of boundary conditions, the type and arrangement of the output edits, etc. It is also necessary to specify the timesteps desired in the case of a burnup study, as well as the power level, which can be specified for each timestep. It is obvious that a complete description of all the capabilities of the code escapes the context of this work.

The items present in a normal output for a burnup study are listed below. It is important to keep in mind that it is hard to define a "normal" output from PDQ-7, since most of the output is optional, and is determined by the user.

a). General definition of the problem: type of problem, geometrical size, figure composition, timestep length, etc.

b). Pictures indicating the geometrical arrangement of the problem, along with the specification of all material compositions.

c). Multiplication factor iterations.

d). Fluxes, power generation, absorption rates, multiplication factors, macroscopic cross sections, etc. groupwise and total, for each edit region defined, which may include separate material regions (v.g. each particular fuel assembly) and/or any combination of them, which can be used to obtain batch-wise or core total values.

e). Description of the burnup step. It may be done in one single calculation, or it may include partial timestep renormalizations in order to correct for isotopic variations during the timestep burning time. Isotopic concentrations for each edit region defined by the user are also listed.

f). Information about the file handling for data transfer between the timestep calculations.

APPENDIX B8
TEMPRET CODE DESCRIPTION

TEMPRET(31) is a thermalhydraulics code developed by Mark Miller for the prediction of two-dimensional, steady state thermal conditions in a PWR elementary fuel cell. The code accepts a wide range of operating characteristics of the fuel, and is able of treating temperature dependent thermodynamic properties of the fuel material, the cladding and the coolant. The heat generation rate can be allowed to vary both in the axial and in the radial directions. The code includes some special features such as the possibility of treating either solid or central-voided fuel pins.

Some of the more significant input data to the code are the following:

a). Geometrical description of the problem.

b). Tabular entry of materials properties.

c). Definition of thermodynamic status of coolant at core inlet.

d). Definition of heat generation distribution.

The output from the code includes the following items:

a). Bulk coolant thermodynamic conditions at each axial node.

b). Reynolds number, Prandtl number, and other heat transfer related data for each axial node.

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c). Departure from nucleate boiling (DNB) heat rate and the ratio for the actual heat rate at each axial node.

d). Coolant quality versus axial position.

e). Clad outer and inner temperature, fuel average and peak temperature and specific heat content, versus axial position.

APPENDIX B9
CONCEPT-IV CODE DESCRIPTION

CONCEPT-IV (32) is a code designed for the computation of the capital cost of an electric generating plant using the steam cycle for power generation. The code is able to compute time- and size-dependent costs of the components of the plant based on a historical data base and an escalation prediction model.

The code breaks the cost of the plant into a set of accounts, including the reactor plant (for the case of a nuclear unit), the turbine plant, structures and facilities, engineering and construction, management services, etc. Each of the different agcounts is broken down into labor, materials and equipment costs. A large number of sub-accounts are considered, yielding a highly detailed cost break-down.

The input needed for the code includes items such as the following:

a). Plant type and location.

b). Net electrical capacity of the plant.

c). Date of purchase of steam generating unit (NSSS for the case of a nuclear unit, or equivalent for the case of a fossil-fired unit).

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d). Date of reception of the construction permit.

e). Date of commercial operation.

f). Interest rate.

The output of the code is extremely flexible, and is basically defined by the user. It can be reduced to a short one-page summary of the major accounts, or it can include detailed listings of all the subaccounts. The code provides also for the possibility of printing a total cumulative cash flow curve.

APPENDIX B10
GEM CODE DESCRIPTION

The initials GEM stand for General Economic Model. The GEM (33) code is devised to calculate the costs of the fuel cycle of a nuclear electric power plant. The code is able to calculate the costs for just a batch or a set of batches of fuel, or it may be used for the computation of the fuel costs through the entire life of the plant.

The code takes into account all the steps of the fuel cycle, from mining through spent fuel shipment and storage, including the costs of extraction, enrichment, etc. and the schedules associated with them.

The code computes both the cumulative fuel cost and the batchwise yearly costs. The calculations are done through three distinct types of economic analysis: Cash Flow, Allocated Costs, and Yearly Cash Flow. The three models yield identical results, but they present them in different forms.

The input to the code includes mainly the following items:

a). Uranium prices.

b). Fabrication and Service costs.

c). Feed losses.

d). Plutonium prices.

e). Fuel weights.

f). Number of enrichments and their magnitudes, tails concentration, and other enrichment data.

g). Economic parameters and payment schedules.

h). Escalation information.

The output from the code includes the following items:

a). Batch economic analysis.

b). Yearly batch and case costs.

c). Yearly and cumulative fuel cycle costs.

d). Case cashflow.

e). Allocated cost analysis.

POWERCO CODE DESCRIPTION

The POWERCO (34) code is used for computing the total busbar cost of electric power generation. This includes all the factors that affect the cost of generation: fuel costs (including shipment costs, possible credits for reprocessable fissile materials, ore costs, etc.), capital costs, investment costs, plant life, capacity factors, tax rates, etc.

The code makes use of three different methods of calculating the power cost. The payout method is the most fundamental one, and it requires a trial-and-error approach. The present worth method avoids this problem by translating all costs through the plant life to a single point in time. More popular than the present worth method is the fixed-charge-rate method; although some approximations are made in its application, which render it less rigorous and more empirical.

The input stream to POWERCO includes the following items:

a). Plant Investmebt.

b). Non-fuel working capital.

c). Project life.

d). Design capacity.

e). Depreciable life.

f). Economic parameters.

g). Cost of initial core.

h). Annual expenditures for uranium, fuel fabrication, transportation, reprocessing (if any), spent fuel credits (if any) etc.

The main data output by the code are the following:

a). Power cost tabulation.

b). Payout tabulation.

c). Tax-deductible expenses.

d). Annual income tax calculation.

e). Fixed charge calculations.

f). Constant annual sales income.

```
10 REM *******************************************
20 REM PWR BORON LETDOWN EVALUATION, + K-EFF AND EOC BURNUP.
30 REM *******************************************
40 CLS:CLEAR1000:DEFINTI,J,K
50 REM *******************************************
60 REM DIMENSIONING AND REACTOR CONFIGURATION
70 REM *******************************************
80 PRINT"PROGRAM FOR CRITICALITY, BORON WORTH, BATCH POWER AND"
90 PRINT"CYCLE LIFE EVALUATION OF A MULTI-BATCH PWR":INPUT"DATA
INPUT FROM (K)EYBOARD OR (D)ISK";A$:IFA$="D"THEN2190
100 CLS:INPUT"TOTAL # OF TIMESTEPS AND EFFECTIVE # OF TIMESTEPS PER
CYCLE";TS,CT:IFCT>TSTHEN100
110 INPUT"NUMBER OF BORON WORTH EVALUATION POINTS";BE:IFBE<>0THEN140
120 INPUT"CONSTANT BORON WORTH IN %K/100PPM, ORJUST
ENTER";BW:BW=BW*1.0E-4:IFBW=0THEN BW=-1.0E-4
130 GOTO210
140 INPUT"# OF FAKE STEPS TO BE KILLED AFTER BORON-WORTH
EVALUATION";CA:IFCA=0THEN160ELSEDIMBK(CA)
150 FORI=1TOCA:INPUT"STEP # TO BE KILLED";BK(I):NEXTI
160 BE=BE*2:DIMBW(BE)
170 REM *******************************************
```

```
180 REM INPUTTING VECTOR OF T-STEPS FOR B-WORTH EVALUATION

190 PRINT"INPUT THE PAIRS OF TIME-STEPS USED FOR B-WORTH EVALUATION"

200 FORI=1TOBE/2:PRINT"PAIR # "I" =";:INPUTBW(2*I-1),BW(2*I):NEXTI

210 INPUT"# OF BATCHES IN THE CORE (MAX=5)";NB:IFNB>5 OR NB<1THEN210

220 DIMR(TS,12),RC(CT,8),RR(NB,2):INPUT"POWER LEVEL OF THE REACTOR
IN MW THERMAL";PL

230 INPUT"FUEL LOADING OF THE CORE IN MTU
(DEFAULT=94.18)";FL:IFFL=0THEN FL=94.18

240 REM ******************************************

250 REM BATCH CONFIGURATION AND OUTER BATCH FLAGGING

260 BN=0

270 FORI=1TOCT:BB=0

280 FORJ=1TONB:PRINT"STEP # THAT FORMS BATCH "J" AT CYCLE STEP
"I;:INPUT RC(I,2+J):NEXTJ

290 INPUT"# OF THE STEP FORMING THE PERIPHERAL BATCH IN THIS
CYCLE-STEP";RC(I,8)

300 FORK=3TO7:IFRC(I,8)=RC(I,K)THENBB=K:NEXTK

310 IFBB<>0THEN330

320 PRINT"THIS PERIPHERAL BATCH DOES NOT BELONG TO THIS
TIME-STEP!!":GOTO290

330 IFBN=0THEN360

340 IF(K-2)=BNTHEN360

350 PRINT"     WARNING !!!":PRINT"PERIPHERAL BATCH IN THIS TIME
STEP DOES NOT BELONG TO              PERIPHERAL BATCH OF PREVIOUS
STEP. RECHECK.":GOTO290

360 BN=K-2:NEXTI
```

```
370 REM ******************************************
380 REM LBP'S WORTH AT BOC AND EOC
390 BB=0:BN=0:BP=0:INPUT"ENTER # OF BATCH (1 IS THE NEWEST)
CONTAINING BURNABLE POISON.    TYPE 'ENTER' IF NONE
DOES.";BP:IFBP=0THEN410
400 INPUT"BATCHWISE REACTIVITY WORTH OF POISON RODS (%K/K) AT BOC &
EOC";BB,BN:IFBN>BBTHEN400
410 FORI=1TOCT:RC(I,2)=0:NEXTI
420 REM ******************************************
430 REM INPUT OF ALL DATA IN 'R' MATRIX
440 CLS:PRINT"INPUT ENDING TIMES (IN HOURS FROM BOL) OF EACH
TIMESTEP"
450 FORI=1TOTS:PRINT"ENDING TIME OF TIMESTEP "I;:INPUTR(I,4):NEXTI
460 CLS:PRINT"INPUT BORON PPM FOR EACH TIMESTEP"
470 FORI=1TOTS:PRINT"BORON PPM OF TIMESTEP "I;:INPUTR(I,6):NEXTI
480 CLS:PRINT"INPUT RELATIVE POWER FOR EACH TIMESTEP"
490 FORI=1TOTS:PRINT"RELPWR OF TIMESTEP "I;:INPUTR(I,2):NEXTI
500 CLS:PRINT"INPUT K-INF FOR EACH TIMESTEP"
510 FOR I=1TOTS
520 PRINT"TIMESTEP "I"  K-INF=";:INPUTR(I,1):IFR(I,1)<3THEN540
530 PRINT"THIS K-INF LOOKS PRETTY WILD !!!!":GOTO520
540 NEXTI
550 REM ***************************************
560 REM CHECKING INPUT
570 REM ***************************************
580 CLS:PRINT"INPUT RELATIVE FLUXES FOR EACH TIMESTEP"
```

```
590 FORI=1TOTS:PRINT"FLUX OF TIMESTEP"I;:INPUTR(I,3):NEXTI
600 CLS:PRINT"CHECK TIMESTEP ENDING TIMES. IF ANY ONE TO BE CHANGED,
INPUT       TIMESTEP NUMBER. OTHERWISE JUST PRESS 'ENTER'":PRINT" "
610 FORI=1TOTS:PRINT"TSTEP "I" ENDS AT "R(I,4)" HR",:NEXTI
620 CH=0:INPUT"ANY CHANGED STEP # ";CH:IFCH=0THEN640
630 INPUT"NEW VALUE";R(CH,4):GOTO600
640 CLS:PRINT"CHECK BORON PPM FOR EACH STEP. ENTER STEP TO BE
CHANGED, OR 0":PRINT" "
650 FORI=1TOTS:PRINT"TIMESTEP "I" HAS "R(I,6)" PPM",:NEXTI
660 INPUT"ANY CHANGED STEP #";CH:IFCH=0THEN680
670 INPUT"NEW VALUE";R(CH,6):CH=0:GOTO640
680 CLS:PRINT"CHECK K FOR EACH TIMESTEP":PRINT"
":PRINT"TSTP";TAB(6)"K-INF";TAB(30)"TSTP";TAB(36)"K-INF"
690
FORI=2TOTSSTEP2:PRINTI-1,TAB(5)R(I-1,1);TAB(30)I;TAB(35)R(I,1):NEXTI
700 CH=0:INPUT"ANY CHANGED STEP";CH:IFCH=0THEN720
710 INPUT"NEW K,";R(CH,1):GOTO680
720 CLS:PRINT"CHECK RELATIVE POWER FOR EACH STEP. ENTER STEP TO BE
CHANGED, OR 0"
730 FORI=1TOTS:PRINT"TIMESTEP "I" HAS RELPOWR="R(I,2),:NEXTI
740 INPUT"ANY CHANGED STEP";CH:IFCH=0THEN760
750 INPUT"NEW VALUE";R(CH,2):CH=0:GOTO720
760 CLS:PRINT"CHECK CORE BATCH COMPOSITION IN CYCLE.":PRINT"ENTER
STEP AND COLUMN TO BE CHANGED OR 'ENTER' TWICE.":A$="########"
770 PRINT"     STEP BATCH-1 BATCH-2 BATCH-3 BATCH-4 BATCH-5
OUTER-BATCH"
```

```
780 FORI=1TOCT:PRINT USING
A$;I;RC(I,3);RC(I,4);RC(I,5);RC(I,6);RC(I,7);RC(I,8):NEXTI
790 INPUT"ANY CHANGED STEP AND COLUMN";CH,CC:IFCH=0THEN810
800 CC=CC+2:INPUT"NEW VALUE";RC(CH,CC):CH=0:GOTO760
810 CC=0:CLS:PRINT"CHECK BATCH WITH LBP AND LBP'S BATCHWISE WORTH
(%K/K) AT BOC,EOC":PRINT"BATCH"BP,"BOC %K/K="BB,"EOC %K/K="BN
820 INPUT"NEW VALUES FOR ALL, OR HIT 'ENTER'";CC,CH,A:IFCC=0THEN840
830 BP=CC:BB=CH:BN=A:GOTO810
840 CC=0:CLS:PRINT"CHECK POWER LEVEL AND FUEL
LOADING":PRINT"POWER="PL"MW THERMAL","FUEL LOADING="FL"MTU"
850 INPUT"NEW VALUES FOR BOTH, OR HIT 'ENTER'";CC,CH:IFCC=0THEN870
860 PL=CC:FL=CH:GOTO840
870 CLS:PRINT"CHECK BORON WORTH EVALUATION POINTS":IFBE<>0THEN900
880 PRINT"NO BORON WORTH EVALUATION POINTS. BORON
WORTH="BW:INPUT"NEW VALUE, OR JUST 'ENTER'";CH:IFCH=0THEN980
890 BW=CH:CH=0:GOTO870
900 IFCA=0THEN930ELSEPRINT"STEPS TO BE KILLED AFTER B-WORTH
EVALUATION (IN INCREASING ORDER) :"
910 FORI=1TOCA:PRINTBK(I),;:NEXTI
920 PRINT" ":PRINT" "
930 PRINT"TIMESTEPS FOR EVALUATION :"
940 FORI=1TOBE/2:PRINTBW(I*2-1),BW(I*2):NEXTI
950 INPUT"WISH TO INPUT A NEW SET (1), OR NO (0)";CH:IFCH=0THEN980
960 FORI=1TOBE/2:PRINT"PAIR #"I:INPUTBW(I*2-1),BW(I*2):NEXTI
970 CH=0:GOTO870
980 REM *************************************
```

```
990 REM STORAGE IN DISK

1000 GOSUB2480

1010 REM ****************************************

1020 REM INITIALIZING WORK MATRIX

1030 RB=10:RP=2

1040 FORI=1TOTS:R(I,8)=R(I,2):R(I,9)=R(I,4):R(I,7)=R(I,6):NEXTI

1050 GOSUB2380

1060 FORI=1TONB:RR(I,1)=RR(I,0):RR(I,2)=RR(I,0):NEXTI

1070 RB=12:OC=R(RC(CT,3),4)

1080 FORI=1TOTS:R(I,12)=R(I,10):NEXTI

1090 REM BORON WORTH VS. CYCLE. INTERPOLATED TO MAKE BORON WORTH AS

A FUNCTION OF PPM

1100 IFBE>0THEN1120

1110 Y=BW:GOTO1170

1120 DEF FNMP(Y1,Y2)=(R(Y1,6)+R(Y2,6))/2

1130 DEF FNWO(Y1,Y2)=(R(Y1,1)-R(Y2,1))/(R(Y1,6)-R(Y2,6))

1140 IFBE>2THEN1190

1150 Y=FNWO(BW(1),BW(2))

1160 FORI=1TOCT:RC(I,1)=Y:NEXTI

1170 WA=Y:WB=0

1180 GOTO1250

1190 XY=0:SX=0:SY=0:XX=0:PRINT"CALCULATING REGRESION LINE OF BORON

WORTH VS. PPM"

1200 FORI=1TOBE/2:X=FNMP(BW(2*I-1),BW(2*I))

1210 Y=FNWO(BW(2*I-1),BW(2*I))

1220 XY=XY+X*Y:SX=SX+X:SY=SY+Y:XX=XX+X*X:NEXTI
```

```
1230 WB=(XY-(SX*SY/(BE/2)))/(XX-(SX*SX/(BE/2))):WA=(SY-WB*SX)/(BE/2)

1240 REM ******************************************

1250 REM WORTH (DELTA K/PPM)=WA+WB*PPM

1260 IFCA=0THEN1270ELSEGOSUB3170

1270 FORI=1TOCT:CH=RC(I,3):RC(I,1)=WA+WB*R(CH,6):NEXTI

1280 REM CALCULATING BORON-FREE K-INF FOR CORRELATION WITH BU.

1290 FORI=1TOTS:R(I-1,11)=R(I,1)-(WA*R(I,6))-(WB*(R(I,6)[2)/2):NEXTI

1300

R(TS,11)=R(TS-1,11)+(R(TS,10)-R(TS-1,10))*(R(TS-1,11)-R(TS-2,11))/(R(

TS-1,10)-R(TS-2,10))

1310 REM ******************************************

1320 REM BEGINNING OF CORE K-EFF CALCULATION

1330 RP=8:GOSUB2380

1340 GOSUB2810

1350 IFBP=0THEN1390

1360 REM CALC. OF LBP-CAUSED REACTIVITY DEFFECT

1370

FORI=1TOCT:CH=RC(I,BP+2):CO=RC(1,BP+2):R(CH,0)=R(CH,0)-BN/100-(BB-BN)

*(R(CH,12)-R(CC,12))/(100*(R(RC(CT,BP+2),10)-R(CC,10))):NEXTI

1380 REM ********************************************

1390 REM MODIFICATION OF K FOR THE EXTERNAL BATCH

1400 GOSUB2680

1410 REM *********************************************

1420 REM OBTAINING BATCH RELATIVE POWERS AND CHECKING FOR DIFFERENCE

WITH INITIALS

1430 TH=2.0
```

```
1440 FORI=1TOCT:SX=0

1450 FORJ=1TONB:CH=RC(I,J+2):IFCH=RC(I,8)THENPB=RC(I,2)ELSEPB=1

1460 SX=SX+((R(CH,0)*PB)[TH]):NEXTJ

1470 FORJ=1TONB:CH=RC(I,J+2):IFCH=RC(I,8)THENPB=RC(I,2)ELSEPB=1

1480 R(CH,5)=((R(CH,0)*PB)[TH]*NB/SX:NEXTJ

1490 NEXTI

1500 MR=0

1510 FORI=1TOTS:IFR(I,5)=0THENNEXTI

1520 CH=ABS((R(I,5)-R(I,2))/(R(I,2)/100)):IFCH>MRTHENMR=CH

1530 NEXTI

1540 REM *********************************************

1550 REM OBTAINING CORE K FOR EACH REAL TIMESTEP AND CHECKING   FOR
DEVIATION

1560 MK=0

1570 FORI=1TOCT:SX=0

1580 FORJ=1TONB:CH=RC(I,J+2):SX=SX+(R(CH,5)/R(CH,0)):NEXTJ

1590 CC=NB/SX:RC(I,0)=CC:IF(ABS(CC-1))>MKTHENMK=ABS(CC-1)

1600 NEXTI

1610 GOSUB3330

1620 REM *********************************************

1630 REM GETTING BORON LETDOWN CURVE

1640 FORI=1TOCT:CC=(1-RC(I,0))/(WA+WB*R(RC(I,3),7))

1650 FORJ=1TONB:CH=RC(I,J+2):R(CH,7)=R(CH,7)+CC:NEXTJ,I

1660 REM *********************************************

1670 REM OBTAINING END OF CYCLE TIME

1680 PRINT"OBTAINING EOC TIME BY INTERPOLATION FOR 0 PPM, K=1"
```

```
1690
     I=CT:XY=0:XX=0:SX=0:SY=0:KK=CT-2:J=0:IFCT=2THENKK=1ELSEIFCT<2THEN1760

1700 CH=RC(I,3):X=R(CH,7):IFI=0THEN1740ELSECC=RC(I-1,3):Y=R(CC,9)

1710 IFCC<=1THEN1720ELSEIFY=R(RC(I-2,3),9)THENKK=KK-1:GOTO1730

1720 XY=XY+X*Y:SX=SX+X:SY=SY+Y:XX=XX+X*X:J=J+1

1730 I=I-1:IFI>=KKTHEN1700

1740 B=(XY-(SX*SY/J))/(XX-(SX*SX/J))

1750 A=(SY-B*SX)/J:GOTO1770

1760 A=R(1,9)

1770 REM A IS NOW THE EOC TIME

1780 A=(A+OC)/2:OC=A

1790 PRINT"END OF CYCLE TIME="A

1800 GOTO3030

1810 REM **************************************

1820 REM **************************************

1830 REM OUTPUT

1840 REM **************************************

1850 INPUT"INPUT A TITLE FOR THE CASE";A$

1860 INPUT"TURN PRINTER ON, THEN ENTER";X

1870 LPRINTA$:LPRINT" ":LPRINT" ":A$="#########":OK=0

1880 FORI=1TONB:LPRINT"BATCH"I:B$="######.##":LPRINT"STEP #
          ";

1890 FORJ=1TOCT:K=RC(J,I+2):LPRINT USING A$;K;:NEXTJ

1900 LPRINT" ":LPRINT"INITIAL STEPS ENDING TIMES";

1910 FORJ=1TOCT:K=RC(J,I+2):LPRINT USING B$;R(K,4);:NEXTJ:GOTO1960
```

```
1920 LPRINT" ":LPRINT"NEW STEPS ENDING TIMES      ";
1930 FORJ=1TOCT:K=RC(J,I+2):LPRINT USING B$;R(K,9);:NEXTIJ
1940 LPRINT" ":LPRINT"NEW TIMESTEP DURATION       ";
1950 FORJ=1TOCT:K=RC(J,I+2):X=R(K,9)-OK:LPRINT USING
B$;X;:OK=R(K,9):NEXTIJ:B$="###.#####":GOTO1980
1960 B$="###.#####":LPRINT" ":LPRINT"INITIAL STEPS K-INF.     ";
1970 FORJ=1TOCT:K=RC(J,I+2):LPRINT USING B$;R(K,1);:NEXTIJ:GOTO2000
1980 LPRINT" ":LPRINT"REACTIVITY-CORRECTED K-INF";
1990 FORJ=1TOCT:K=RC(J,I+2):LPRINT USING B$;R(K,0);:NEXTIJ:GOTO2040
2000 LPRINT" ":LPRINT"CLEAN K-INF.             ";
2010 FORJ=1TOCT:K=RC(J,I+2):LPRINT USING B$;R(K-1,11);:NEXTIJ
2020 LPRINT" ":LPRINT"INITIAL BORON PPM.        ";
2030 FORJ=1TOCT:K=RC(J,I+2):LPRINT USING A$;R(K,6);:NEXTIJ:GOTO2060
2040 LPRINT" ":LPRINT"NEW BORON PPM.            ";
2050 FORJ=1TOCT:K=RC(J,I+2):LPRINT USING A$;R(K,7);:NEXTIJ:GOTO2080
2060 LPRINT" ":LPRINT"INITIAL RELATIVE POWERS   ";
2070 FORJ=1TOCT:K=RC(J,I+2):LPRINT USING
B$;R(K,2);:NEXTIJ:B$="######.##":GOTO1920
2080 LPRINT" ":LPRINT"NEW RELATIVE POWERS       ";
2090 FORJ=1TOCT:K=RC(J,I+2):LPRINT USING B$;R(K,5);:NEXTIJ
2100 LPRINT" ":LPRINT" ":LPRINT" "
2110 NEXTI
2120 LPRINT" ":LPRINT" ":X=0
2130 FORI=1TONB:LPRINT"ESTIMATED BURNUP FOR
CYCLE"I"IS"RR(I,0)"MWD/MTU":X=X+RR(I,0):NEXTI
2140 K=RC(CT,3):LPRINT"END OF LIFE BURNUP IS ESTIMATED
```

```
AT"X"MWD/MTU":LPRINT"CYCLE LENGTH IS ESTIMATED AS"A"HOURS, WHILE
INPUT ESTIMATE WAS"R(K,4)"HOURS"
2150 LPRINT"MAXIMUM PERCENT CHANGE BETWEEN OLD AND NEW ESTIMATES OF
STEP RELATIVE POWERS IS"MR
2160 LPRINT"BATCHWISE BORON WORTH REGRESSION LINE IS:
W="WA"+("WB"*PPM)"
2170 STOP
2180 REM **************************************
2190 REM ROUTINE FOR DATA INPUT FROM DISK
2200 INPUT"FILESPEC";A$
2210 OPEN"R",1,A$:FIELD 1,2 AS X1$,2 AS X2$,2 AS X3$,2 AS X4$,4 AS
X5$,4 AS X6$,2 AS X7$,4 AS X8$,4 AS X9$,229 AS N$
2220 GET
1:TS=CVI(X1$):CT=CVI(X2$):NB=CVI(X3$):BE=CVI(X4$):PL=CVS(X5$):FL=CVS(
X6$):A$=N$:BP=CVI(X7$):BB=CVS(X8$):BN=CVS(X9$)
2230
CC=201:CH=1:I=1:K=1:DIMR(TS,12),RC(CT,8),RR(NB,2):IFBE<>0THEN2250
2240 BW=-1.0E-4:GOTO2260
2250 DIMBW(BE)
2260 FORJ=3TO8:RC(I,J)=CVS(MID$(A$,K,4)):K=K+4:NEXTJ
2270 I=I+1:IFK>CCTHEN2290
2280 IF I<=CTTHEN2260ELSE2300
2290 GET 1:CC=223:FIELD 1,255 AS N$:K=1:A$=N$:GOTO2280
2300 IFCH>TSTHEN2330
2310 FORJ=1TO6:R(CH,J)=CVS(MID$(A$,K,4)):K=K+4:NEXTJ
2320 CH=CH+1:GOTO2270
```

```
2330 IFBE=0THEN2370

2340 FORJ=1TOBE:BW(J)=CVI(MID$(A$,K,2)):K=K+2:NEXTJ

2350 CA=CVI(MID$(A$,K,2)):K=K+2:IFCA=0THEN2370ELSEDIMBK(CA)

2360 FORJ=1TOCA:BK(J)=CVI(MID$(A$,K,2)):K=K+2:NEXTJ

2370 CLOSE:GOTO560

2380 REM OBTAINING EOC BURNUPS (CYCLE AND STEPWISE)

2390 X=0:CC=1

2400
FORI=1TOTS:X=X+PL*R(I,RP)*(R(I,9)-R(I-1,9))/(24*FL):R(I,RB)=X:IFI=RC(
CT,CC+2)THEN2410ELSENEXTI

2410 CH=0

2420 FORJ=0TOCC-1:CH=CH+RR(J,0):NEXTJ

2430 RR(CC,0)=X-CH:CC=CC+1:IFI<TSTHENNEXTI

2440 GOSUB3270

2450 RETURN

2460 REM THE FINAL BURNUP OF EACH CYCLE I HAS BEEN PLACED ON RR(I,0)

2470 REM ************************************

2480 REM ROUTINE FOR STORAGE OF DATA IN DISK

2490 CLS:INPUT"DO YOU WANT TO SAVE DATA ON DISK? (Y)ES OR
(N)O";A$:IFLEFT$(A$,1)<>"Y"THENRETURN

2500 INPUT"FILESPEC";A$:OPEN"R",1,A$:FIELD 1,2 AS X1$,2 AS X2$,2 AS
X3$,2 AS X4$,4 AS X5$,4 AS X6$,2 AS X7$,4 AS X8$,4 AS X9$,229 AS N$

2510 LSET X1$=MKI$(TS):LSET X2$=MKI$(CT):LSET X3$=MKI$(NB):LSET
X4$=MKI$(BE):LSET X5$=MKS$(PL):LSET X6$=MKS$(FL):LSET
X7$=MKI$(BP):LSET X8$=MKS$(BB):LSET X9$=MKS$(BN)

2520 A$="":CO=201:CH=1:I=1
```

```
2530 FORJ=3TO8:A$=A$+MKS$(RC(I,J)):NEXTJ

2540 I=I+1:IFLEN(A$)>CCTHEN2560

2550 IFI<=CTTHEN2530ELSE2570

2560 LSETN$=A$:PUT 1:A$="":CC=223:FIELD 1,255 AS N$:GOTO2550

2570 IFCH>TSTHEN2600

2580 FORJ=1TO6:A$=A$+MKS$(R(CH,J)):NEXTJ

2590 CH=CH+1:GOTO2540

2600 IFBE=0THEN2640

2610 FORJ=1TOBE:A$=A$+MKI$(BW(J)):NEXTJ

2620 A$=A$+MKI$(CA):IFCA=0THEN2640

2630 FORJ=1TOCA:A$=A$+MKI$(BK(J)):NEXTJ

2640 IFA$=""THEN2660

2650 LSETN$=A$:PUT 1

2660 CLOSE:RETURN

2670 REM *************************************

2680 REM ROUTINE FOR CALCULATION OF MODIFICATION OF K FOR THE
EXTERNAL BATCH

2690 FORI=1TOCT:CH=RC(I,8):TA=-20.0:BU=R(CH-1,12)

2700 L2=(0.486-(1.9E-7*BU))/(0.1912+(6.42E-7*BU))

2710
SA=(0.486-(1.9E-7*BU))/(SQR((0.4033-(5.135E-6*R(CH,7)))*(0.0166+(2.72
7E-5*R(CH,7)))))

2720 B2=(2.405/(168.53+SA))[2:UB=0

2730 REM SHAPE FACTOR CALCULATION

2740 FORJ=1TONB:UH=RC(I,J+2):UB=UB+R(UH,3):NEXTJ

2750
```

```
Z1=2.405*SQR((NB-1)/NB):Z1=(Z1/2)-((Z1[3]/16)+((Z1[5]/384):SG=(1/(1-Z

1*1.84292*SQR((NB-1)/NB))))[2:SF=SG*R(CH,3)/UB

2760 PR=EXP(TA*B2)/(1+L2*B2)

2770 PB=1-((1-PR)*SF*0.7)

2780 R(CH,0)=R(CH,0)*PB:RC(I,2)=PB:PRINT"MODIF FOR EXTERNAL BATCH

T-STEP"I"="PB

2790 NEXTI

2800 RETURN

2810 REM CALCULATING WORK K-INF FROM PRESENT-STATUS BURNUPS AND

BORON CONCENTRATIONS.

2820 R(1,0)=R(0,11)+(WA*R(1,7))+(WB*(R(1,7)[2])/2):IFTS<2THENRETURN

2830 FORI=2TOTS:X=R(I-1,12):XL=0:XM=0

2840 FORJ=0TOTS:IFR(J,10)<XTHENXL=J

2850 IFR(J,10)>=XTHENXM=J

2860 IFXM=J AND J<>0THENJ=TS

2870 NEXTJ

2880 IFXL=XMTHENPRINT"WARNING!! XM=XL !!!":STOP

2890 IFXM=0 AND XL<>0 THEN XM=TS:XL=TS-2

2900 IFABS(R(XL,11)-R(XM,11))<0.001 AND XL>1THENXL=XL-1:GOTO2900

2910

R(I,0)=R(XL,11)+((X-R(XL,10))*(R(XM,11)-R(XL,11))/(R(XM,10)-R(XL,10))

):R(I,0)=R(I,0)+(WA*R(I,7))+(WB*(R(I,7)[2])/2):NEXTI

2920 RETURN

2930 REM CHECKING FOR CONSISTENCY OF CYCLES' BU

2940 CLS:PRINTTAB(15);"K CONVERGED. CHECKING BURNUPS."

2950 XX=0:RP=5:GOSUB2380
```

```
2960 FORI=1TONB:CH=ABS((RR(I,0)-RR(I,2))/RR(I,2)):IFCH>XXTHENXX=CH

2970 NEXTI

2980 IFXX<=0.01THEN1850

2990 FORI=1TOTS:R(I,8)=(R(I,5)+R(I,8))/2:NEXTI

3000 RP=8:GOSUB2380

3010 FORI=1TONB:RR(I,2)=RR(I,0):NEXTI

3020 GOTO1340

3030 REM ASSIGNING NEW STEP ENDING TIMES

3040 X=0:Y=R(RC(CT,3),9)

3050 FORJ=2TONB:CH=RC(CT,J+1)+1:CC=RC(CT,J+2)

3060 FORI=CHTOCC:R(I,9)=R(I,9)+(A-Y)*(J-1):NEXTI,J

3070 FORI=1TONB

3080 FORJ=1TOCT:CH=RC(J,I+2):Y=R(CH,9):IFY>=A+XTHEN3100

3090 IFJ=CTTHEN3100ELSEIFY<R(CH+1,9)THENNEXTJELSER(CH+1,9)=X+A:NEXTJ

3100 R(CH,9)=X+A:NEXTJ

3110 X=X+A:NEXTI

3120 FORI=1TONB:CC=CT

3130 CH=RC(CC,I+2):IFR(CH-1,9)<>A*ITHEN3150

3140 R(CH,7)=0:CC=CC-1:IFCC=0THEN3150ELSE3130

3150 NEXTI

3160 IFMK<0.01THEN2930ELSE1330

3170 FORI=CATO1STEP-1

3180 FORJ=BK(I)TOTS-1

3190 FORK=0TO12:R(J,K)=R(J+1,K):NEXTK,J,I

3200 TS=TS-CA

3210 FORI=1TOCT
```

```
3220 FORJ=3TONB+2
3230 FORK=CATO1STEP-1:IFRC(I,J)>BK(K)THENRC(I,J)=RC(I,J)-K:GOTO3250
3240 NEXTK
3250 NEXTJ,I
3260 RETURN
3270 CLS
3280 PRINT"CYCLES' BURNUPS MATRIX"
3290 FORI=1TONB
3300 FORJ=0TO2:PRINTRR(I,J),;:NEXTJ
3310 PRINT" ":NEXTI
3320 RETURN
3330 CLS:PRINT"STEP","END TIME","WORK K","CORE K"
3340 FORI=1TOTS:PRINTI,R(I,9),R(I,0),;
3350 IFI<=CTTHENPRINTRC(I,0)ELSEPRINT" "
3360 NEXTI
3370 RETURN
```

APPENDIX C2
SAMPLE RUN OF CRIBUR

Cribur was first developed for use on a microcomputer, and for ease of operation, it was designed for interactive use. Therefore, all information to and from the computer is on the CRT screen, except the final output, which is printed on hard copy. The following listing is a reproduction of the screen messages that appear on a sample run of the code, and of the final printout. During execution, the code prints sets of data on the screen that indicate the progress of the calculation and the convergence of the iterations. These data are not presented here.

Screen 1:

PROGRAM FOR CRITICALITY, BORON WORTH, BATCH POWER
AND CYCLE LIFE EVALUATION OF A MULTI-BATCH PWR.

DATA INPUT FROM (K)EYBOARD OR (D)ISK ? D
FILESPEC ? HP50A252

Screen 2:

CHECK TIMESTEP ENDING TIMES. IF ANY ONE TO BE CHANGED, INPUT
TIMESTEP NUMBER. OTHERWISE, JUST PRESS "ENTER".

TSTEP 1 ENDS AT   200 HR   TSTEP 2 ENDS AT 2000 HR
TSTEP 3 ENDS AT 6000 HR   TSTEP 4 ENDS AT 12000 HR

293

TSTEP 5 ENDS AT 16300 HR   TSTEP 6 ENDS AT 16500 HR

TSTEP 7 ENDS AT 18300 HR   TSTEP 8 ENDS AT 22300 HR

TSTEP 9 ENDS AT 22301 HR   TSTEP 10 ENDS AT 28300 HR

TSTEP 11 ENDS AT 28301 HR   TSTEP 12 ENDS AT 32600 HR

TSTEP 13 ENDS AT 32800 HR   TSTEP 14 ENDS AT 34600 HR

TSTEP 15 ENDS AT 38600 HR   TSTEP 16 ENDS AT 44600 HR

TSTEP 17 ENDS AT 48900 HR   ANY CHANGED STEP # ?

    Screen 3:

CHECK BORON PPM FOR EACH STEP. ENTER STEP TO BE CHANGED, OR "ENTER"

TIMESTEP 1 HAS 1310 PPM   TIMESTEP 2 HAS 1170 PPM

TIMESTEP 3 HAS 1040 PPM   TIMESTEP 4 HAS 750 PPM

TIMESTEP 5 HAS 310 PPM   TIMESTEP 6 HAS 1310 PPM

TIMESTEP 7 HAS 1170 PPM   TIMESTEP 8 HAS 1040 PPM

TIMESTEP 9 HAS 650 PPM   TIMESTEP 10 HAS 750 PPM

TIMESTEP 11 HAS 150 PPM   TIMESTEP 12 HAS 310 PPM

TIMESTEP 13 HAS 1310 PPM   TIMESTEP 14 HAS 1170 PPM

TIMESTEP 15 HAS 1040 PPM   TIMESTEP 16 HAS 750 PPM

TIMESTEP 17 HAS 310 PPM   ANY CHANGED STEP # ?

    Screen 4:

CHECK K FOR EACH TIMESTEP

TSTP	K-INF	TSTP	K-INF
1	1.1662	2	1.1489
3	1.1478	4	1.1415
5	1.1320	6	1.0032

7	1.0142	8	1.0137
9	1.0216	10	1.0127
11	1.0288	12	1.0136
13	0.8989	14	0.9103
15	0.9126	16	0.9181
17	0.9276	ANY CHANGED STEP ?	

Screen 5:

CHECK RELATIVE POWER FOR EACH STEP.

ENTER STEP TO BE CHANGED, OR PRESS "ENTER".

TIMESTEP 1 HAS RELPWR= 1.176 TIMESTEP 2 HAS RELPWR= 1.14

TIMESTEP 3 HAS RELPWR= 1.14 TIMESTEP 4 HAS RELPWR= 1.13

TIMESTEP 5 HAS RELPWR= 1.11 TIMESTEP 6 HAS RELPWR= 1.01

TIMESTEP 7 HAS RELPWR= 1.03 TIMESTEP 8 HAS RELPWR= 1.03

TIMESTEP 9 HAS RELPWR= 1.03 TIMESTEP 10 HAS RELPWR= 1.03

TIMESTEP 11 HAS RELPWR= 1.03 TIMESTEP 12 HAS RELPWR= 1.03

TIMESTEP 13 HAS RELPWR= 0.81 TIMESTEP 14 HAS RELPWR= 0.83

TIMESTEP 15 HAS RELPWR= 0.83 TIMESTEP 16 HAS RELPWR= 0.84

TIMESTEP 17 HAS RELPWR= 0.86 ANY CHANGED STEP ?

Screen 6:

CHECK CORE BATCH COMPOSITION IN CYCLE.

ENTER STEP AND COLUMN TO BE CHANGED, OR JUST PRESS "ENTER"

STEP	BATCH-1	BATCH-2	BATCH-3	BATCH-4	BATCH-5	OUTER-BATCH
1	1	6	13	0	0	1
2	2	7	14	0	0	2

3	3	8	15	0	0	3
4	4	10	16	0	0	4
5	5	12	17	0	0	5

ANY CHANGED STEP AND COLUMN ?

Screen 7:

CHECK BATCH WITH LBP AND LBP'S BATCHWISE WORTH (%K/K) AT BOC, EOC

BATCH 0   BOC %K/K= 0   EOC %K/K= 0

NEW VALUES FOR ALL OR JUST PRESS "ENTER" ?

CHECK POWER LEVEL AND FUEL LOADING

POWER= 1700 MW THERMAL   FUEL LOADING= 94.18 MTU

NEW VALUES FOR BOTH OR HIT "ENTER" ?

Screen 8:

CHECK BORON WORTH EVALUATION POINTS

STEPS TO BE KILLED AFTER B-WORTH EVALUATION (IN INCREASING ORDER):

9   11

TIMESTEPS FOR EVALUATION:

6   7

9   10

11   12

WISH TO INPUT A NEW SET Y/N ? N

Screen 9:

DO YOU WANT TO STORE DATA ON DISK? Y/N ? N

Code output reproduction:

HP50A252. 50% POWER. PITCH = 1.25

BATCH 1

STEP #	1	2	3	4	5
INITIAL STEPS ENDING TIME	200.00	2000.00	6000.00	12000.00	16300.00
INITIAL STEPS K-INF.	1.16620	1.14890	1.14780	1.14150	1.13200
CLEAN K-INF.	1.28232	1.25395	1.24228	1.21141	1.16201
INITIAL BORON PPM	1310	1170	1040	750	310
INITIAL RELATIVE POWERS	1.17600	1.14000	1.14000	1.13000	1.11000
NEW STEPS ENDING TIMES	200.00	2000.00	6000.00	12000.00	16529.40
NEW TIMESTEP DURATION	200.00	1800.00	4000.00	6000.00	4529.36
REACTIVITY-CORRECTED K-INF	1.08072	1.06587	1.06482	1.06278	1.05874
NEW BORON PPM	1321	1200	1055	761	331
NEW RELATIVE POWERS	1.05971	1.02967	1.02789	1.02451	1.01790

BATCH 2

STEP #	6	7	8	9	10
INITIAL STEPS ENDING TIME	16500.00	18300.00	22300.00	28300.00	32600.00
INITIAL STEPS K-INF.	1.00320	1.01420	1.01370	1.01270	1.01360
CLEAN K-INF.	1.11932	1.11925	1.10818	1.08261	1.04361
INITIAL BORON PPM	1310	1170	1040	750	310
INITIAL RELATIVE POWERS	1.01000	1.03000	1.03000	1.03000	1.03000
NEW STEPS ENDING TIMES	16729.40	18529.40	22529.40	28529.40	33058.70
NEW TIMESTEP DURATION	200.00	1800.00	4000.00	6000.00	4529.36
REACTIVITY-CORRECTED K-INF	1.01486	1.02252	1.02000	1.01807	1.01601

NEW BORON PPM	1321	1200	1055	761	331
NEW RELATIVE POWERS	1.08646	1.10146	1.09566	1.09068	1.08486

BATCH 3

STEP #	11	12	13	14	15
INITIAL STEPS ENDING TIME	32800.00	34600.00	38600.00	44600.00	48900.00
INITIAL STEPS K-INF.	0.89893	0.91031	0.91261	0.91807	0.92765
CLEAN K-INF.	1.01505	1.01536	1.00709	0.98798	0.95766
INITIAL BORON PPM	1310	1170	1040	750	310
INITIAL RELATIVE POWERS	0.81000	0.83000	0.83000	0.84000	0.86000
NEW STEPS ENDING TIMES	33258.70	35058.70	39058.70	45058.70	49588.10
NEW TIMESTEP DURATION	200.00	1800.00	4000.00	6000.00	4529.36
REACTIVITY-CORRECTED K-INF	0.89967	0.90816	0.91227	0.91697	0.92399
NEW BORON PPM	1321	1200	1055	761	331
NEW RELATIVE POWERS	0.85383	0.86886	0.87645	0.88481	0.89724

ESTIMATED BURNUP FOR CYCLE 1 IS 12736.5 MWD/MTU

ESTIMATED BURNUP FOR CYCLE 2 IS 13568.3 MWD/MTU

ESTIMATED BURNUP FOR CYCLE 3 IS 10990.7 MWD/MTU

END OF LIFE BURNUP IS ESTIMATED AT 37295.5 MWD/MTU

CYCLE LENGTH IS ESTIMATED AS 16529.4 HOURS, WHILE

    INPUT ESTIMATE WAS 16300 HOURS.

MAXIMUM PERCENT CHANGE BETWEEN OLD AND NEW ESTIMATES OF

    STEP RELATIVE POWERS IS 9.88877

BATCHWISE BORON WORTH REGRESSION LINE IS: W=-9.9343E-5+(1.63E-8*PPM)

APPENDIX D
ISOTOPIC AND SPECTRAL DATA FROM BURNUP CALCULATIONS

Chapter IV describes the burnup optimization calculations performed for the standard and the low power density cores, and the main neutronic and plant performance results that can be drawn from them.

Table D.1 shows the cell-homogenized number densities of U-235, Pu-239, Pu-240 and Pu-241 at the beginning and end of life of the fuel pins for all the core cases studied in Chapter IV. All plutonium concentrations are assumed to be zero at BOL, since the initial fuel composition is suposed to contain only uranium isotopes. Note that the BOL concentration of U-235 decreases as the pitch is increased. This is the effect of the cell homogenization of the number density. All fuels are assumed to start with a 3.1% enriched uranium composition. A pellet density of 93% is assigned, which is the default assumed by EPRI-CELL. The number densities shown include also the effect of thermal fuel expansion, as calculated by EPRI-CELL.

Table D.2 lists the cell average neutron velocity for the sixteen cores studied, for their BOL, MOL and EOL. As expected, the cores with larger pitches show a lower velocity, due to the better neutron moderation. The power level for any particular pitch does not show a significant effect on the average velocity.

Fuel burnup affects the neutron velocity, consistently causing a softening of the spectrum as fuel exposure increases.

Table D.1
Cell-Homogenized Number Densities (Atoms/Barn CM X 10^6)

Pitch	% Power	BOL U-235	EOL U-235	EOL Pu-239	EOL Pu-240	EOL Pu-241
1.20	50	258.24	62.08	47.63	19.38	12.07
1.20	33	258.24	59.82	46.97	19.68	11.74
1.20	25	258.24	58.35	46.96	19.87	11.52
1.25	100	238.00	50.44	39.08	17.53	10.57
1.25	50	238.00	47.46	38.97	18.33	10.51
1.25	33	238.00	46.73	38.49	18.36	10.15
1.25	25	238.00	45.79	38.36	18.45	9.92
1.30	50	220.00	39.47	32.47	16.77	8.96
1.30	33	220.00	38.11	32.14	16.86	8.72
1.30	25	220.00	37.44	31.98	16.89	8.51
1.35	50	204.00	32.40	27.59	15.78	7.61
1.35	33	204.00	31.99	27.32	15.51	7.54
1.35	25	204.00	31.46	27.19	15.52	7.36
1.40	50	189.70	29.56	23.50	13.99	6.56
1.40	33	189.70	28.61	23.23	14.06	6.38
1.40	25	189.70	28.15	23.08	14.09	6.22

Table D.2
Average Cell Neutron Velocities (eV)

Pitch	% Power	BOL Speed	MOL Speed	EOL Speed
1.20	50	1.93	1.89	1.87
1.20	33	1.93	1.89	1.87
1.20	25	1.93	1.89	1.87
1.25	100	1.89	1.86	1.83
1.25	50	1.90	1.86	1.83
1.25	33	1.89	1.85	1.83
1.25	25	1.89	1.85	1.83
1.30	50	1.87	1.82	1.80
1.30	33	1.87	1.82	1.80
1.30	25	1.87	1.82	1.79
1.35	50	1.85	1.80	1.78
1.35	33	1.85	1.80	1.77
1.35	25	1.85	1.80	1.77
1.40	50	1.83	1.78	1.75
1.40	33	1.83	1.78	1.75
1.40	25	1.83	1.78	1.75

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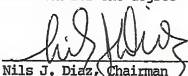
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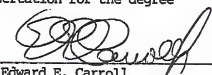
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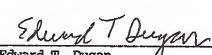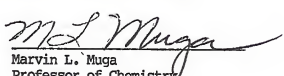
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